COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

PRINCIPAL WORKING GROUP N° 1

SPECIALIST MEETING ON FIRE PROTECTION AND FIRE PROTECTION SYSTEMS IN NUCLEAR POWER PLANTS

Held on December 6th-9th, 1993
in Cologne, Germany

FOR TECHNICAL REASONS, THIS DOCUMENT IS NOT AVAILABLE ON OLIS.
Specialist Meeting
on
FIRE PROTECTION AND FIRE PROTECTION SYSTEMS IN NUCLEAR POWER PLANTS

PROCEEDINGS

Hosted by

Gesellschaft für Anlagen und Reaktorsicherheit (GRS)

held on 6th-9th December, 1993
Cologne, Germany
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (NEA), is an international committee made up of senior scientists and engineers. It was set up in 1973 to develop and coordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee’s purpose is to foster international cooperation in nuclear safety among the OECD Member countries.

The CSNI constitutes a forum for the exchange of technical information and for collaboration between organizations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus on technical issues of common interest. It promotes the coordination of work in different Member Countries including the establishment of cooperative research projects and results to participating organizations. Full use is also made of traditional methods of cooperation, such as information exchanges, establishment of working groups, and organization of conferences and specialist meetings.

The greater part of the CSNI’s current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the nuclear fuel cycle, conducts periodic surveys of the reactor safety research programmes and operates an international mechanism for exchanging reports on safety related nuclear power plant accidents.

In implementing its programme, the CSNI establishes cooperative mechanisms with NEA’s Committee of Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regards to safety. It also cooperates with NEA’s Committee on Radiation Protection and Public Health and NEA’s Radioactive Waste Management Committee on matters of common interest.
Specialist Meeting on Fire Protection and Fire Protection Systems in NPPs

CONTENTS

Welcoming Address, Dr. Kotthoff
NEA Introductory Remarks
Session and Panel Summaries
General Programme
Technical Papers
List of Participants
WELCOMING ADDRESS

Dr. K. Kotthoff, GRS, Germany

On behalf of the Chairman of GRS, who regrets not being able to be with you this morning, I am very pleased to welcome all participants in Cologne at GRS Headquarters to our CSNI Specialist Meeting on Fires and Fire Protection Systems in Nuclear Power Plants. The CSNI is the Committee on the Safety of Nuclear Installations of the Nuclear Energy Agency of the OECD. Our Specialist Meeting has been organized by the CSNI Principal Working Group N° 1 dealing with "Operating Experiences and Human Factors" and I am very glad that we have with us today some members of the Principal Working Group N° 1. Furthermore, I would like to welcome Mr. Clausner from the Nuclear Energy Agency of the OECD who carried a large burden in organizing this meeting. Unfortunately, Mr. Reig, from Spain, the Chairman of Principal Working Group N° 1, was unable to attend the meeting. Therefore, Mr. Clausner will welcome you on behalf of the Nuclear Energy Agency of the OECD.

Our meeting today is one of a series of CSNI Specialist Meetings promoted by the Principal Working Group N° 1 under the auspices of the CSNI. CSNI Specialist Meetings are intended to provide a forum to specialists of the OECD Member countries for the in-depth discussion of specific relevant technical topics.

Traditionally, Specialist Meetings are hosted by Member countries of the NEA. I would like to emphasize that it is an honour for the FRG and especially the GRS to host the CSNI Specialist Meeting of Fires and Fire Protection Systems in NPPs. GRS has been working for many years in the field of fire protection in NPPs as far as nuclear safety is concerned. Thus, we hope to be a good host during this meeting.

The meeting has been organized by a Programme Committee which included nominated members from Belgium, FRG, Japan, Sweden, IAEA and NEA. The Programme Committee met in July in Paris to agree on the practical arrangements for the meeting and the paper selection.

Operating experience from the past and even the recent past point out the high importance of adequate fire protection in NPPs for nuclear safety. Therefore, I appreciate the broad interest in the meeting from regulators and their consultants, utilities, manufacturers, research facilities and international organizations. Due to the importance of the topic, CSNI has decided not to restrict the meeting to specialists from OECD countries and I appreciate that experts from several non-OECD countries took the opportunity to stay with us during this meeting.

I hope that this meeting provides an in-depth exchange of knowledge and experience with respect to fire and fire protection systems in NPPs. I also hope that all participants will benefit from the papers presented and from the discussions which are an essential part of such meetings to promote the exchange of knowledge and experience. Have a very pleasant stay in Cologne, and hopefully you will have some time and good weather to have a look around this interesting city with a history of more than 2000 years.

GRS-601 kot-brv 06.12.1993

1
Introductory remarks by J.-P. Clausner, NEA Secretariat

On behalf of the Nuclear Energy Agency of the OECD, I would like to address a warm welcome to all participants in this Specialist Meeting on Fire Protection and Fire Protection Systems in Nuclear Power Plants and a special thank you to Dr. K. Kothoff of GRS for his kind invitation to host the meeting and to Dr. Röwekamp and Mr. Liemersdorf for organising it here in Cologne.

As you know, this specialist meeting has been organised by the Principal Working Group N° 1 (PWG1) of the Committee on the Safety of Nuclear Installations (CSNI) of the NEA. For those of you who are not familiar with that organisation, I will present a brief overview of both its role and its activities.

The NEA is one of the fifteen bodies that make up the Paris-based Organisation for Economic Cooperation and Development (OECD). Since the majority of the OECD’s activities are oriented towards economics, the role of the Agency is less known and may sometimes be confused with that of the International Atomic Energy Agency (IAEA) in Vienna. In connection with the safety of nuclear power plants, the charter of the NEA calls for "...the promotion of the safety of nuclear installation...", "...the dissemination of information...on the safety and regulation of nuclear activities...".

NEA’s activities in the vast domain of nuclear safety and regulation include: operating experience and human factors, primary coolant system behaviour, reactor component integrity, probabilistic safety assessment, severe accident management, regulatory activities, safety of the nuclear fuel cycle (other installations than reactors), and specific projects (OECD Halden Project, Programme for the Inspection of Steel Components (PISC)). These activities are determined by the two standing Committees which are concerned with the safety of nuclear installations: the Committee on the Safety of Nuclear Installations (CSNI) and the Committee on Nuclear Regulatory Activities (CNRA) which consist of delegates representing the NEA Member countries. The CSNI is responsible for technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of these installations. The CNRA is a forum for the exchange of information among regulatory organisations. Due to the vastness of the technical field of nuclear safety, the CSNI formed working groups to deal with the areas of interest which are mentioned above.

Among these groups, PWG1 deals with issues related to operating experience and human factors. Experts forming the group identify safety issues of common interest and propose solutions.

---

1 The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972 when Japan became its first non-European full member. NEA membership today consists of all European member countries of the OECD as well as Australia, Canada, Japan, Korea and the United States.
to be brought to the attention to regulators and utilities. The work of the group is centered on the Incident Reporting System (IRS) created by the NEA in 1980 with the objective of collecting reports of incidents of safety significance and disseminating lessons learned for the benefit of the international nuclear community.

Among the 2000 reports which are currently stored in the IRS database, PWG1 identified a number of significant events involving fires and fire protection systems. Some events raised concerns regarding fires that could constitute a potential element of common cause failure and their possible consequences on plant safety. Fire hazard analysis were conducted in Member countries and PWG1 specialists agreed on exchanging information, on the international scene, on measures taken or to be taken from the design stage to the operating stage to ensure that essential factors have been considered in every aspect of fire prevention, detection and fighting in order to verify that fire protection in NPPs responds to the following main objectives:

- ensure performance of safety functions,
- ensure security of plant personnel,
- limit equipment damages likely to result in extended outages.

This 4-day meeting has been divided in five technical sessions aimed at providing a forum for exchanging information of current issues in fire protection principles, operating experience, regulatory requirements, risk assessment, research and development, and hopefully, developing conclusions regarding these issues.

I have no doubt that with such assistance this objective will be met and I now wish you fruitful discussions and a successful meeting.
Summary Session 1

This session contained 4 papers.

Mr. Zhong from the IAEA presented the programme presently being developed by the Agency in the matter of fire protection inspection guidelines. This programme should be completed in 1995 and will include the establishment of a safety guide providing guidelines for the inspection of the entire inspection programme, from fire hazard analysis to the implementation of fire protection techniques. Efforts are also made to develop guidelines on performing a fire hazard analysis, on PSA-based fire risk assessment and on assessment of fire safety.

The use of fast response sprinklers was discussed by Mr. Arvidson, based on a paper written by Mr. Ingason from the Swedish National Testing and Research Institute. The use of this type of sprinklers is suggested to replace the standard models due to their ability to work quicker in case of fire, giving a better chance to control and/or suppress the fire. The classification of sprinkler heads by use of the Response Time Index (RTI) parameter has been discussed. The use of fast response sprinklers in replacement of deluge systems should however be considered with care.

Messrs. Andersson and Bergkvist from the FORSMARKS power plant informed us of the philosophy adopted concerning fire prevention and fire protection in these plants. They discussed the problems connected with the use of a sophisticated computerized fire alarm system. This system had to be upgraded due to bad initial design of the computer program and due to ineffective hardware construction between the computer systems. After the upgrading of the central fire alarms and the addition of a complementary supervisory computer system, the objective of detection of the design base fire could be obtained. Two hundred and fifty (250) detectors were activated in a very short period of time (20 minutes) with successful results.

Mr. Arvidson presented the research mode of water mist extinguishing systems at the Swedish National Testing and Research Institute. Water mist is very effective in heat reduction in a fire. Getting the water on the burning surface in order to extinguish the fire remains a key problem. This technology requires additional basic research before becoming fully reliable and operational.

A. Vandewalle - Chairman of Session 1
Summary Session 2

This session contained 5 papers: Two from authorities
One from a utility
One from a vendor
One from a research institute

Authorities

Mr. Madden (NRC), described how the concept "Defense-in-depth" has been used at the NRC in the regulatory work for assuring post-fire safe-shutdown capability. The following principles have been used to create a balanced approach:

1. Preventing fires from starting;
2. Promptly detecting fires in any location of the plant, controlling and rapidly suppressing the fires that do occur and limiting the damage;
3. Designing plant safety systems so that even if a fire were to propagate within a plant area, the fire will not prevent essential plant functions from being performed.

To fulfill these principles, the NRC has established fire protection guidelines to implement operational fire protection features in the design of the facilities. Generally, plant fire protection features include installing full automatic detection; limiting fire propagation by compartmentalizing the plant into fire areas and providing fire resistance barrier systems for criteria, safe shutdown compartments.

Mr. Cowley (NII) presented the United Kingdom approach to fire hazard analysis. The basis for this approach is the revised Safety Assessment Principles for Nuclear Power Plants from 1992, which includes a general requirement that adequate fire detection and fire fighting systems should be provided.

A fire hazard analysis should be carried out to determine the potential effects on safety systems.

Mr. Cowley described the methods adopted in assessing the adequacy of the protection against fire on plants in construction and on future designs. He also described the work being done with fire provisions on the older UK magnox reactors.

Utility

From the utility side, Mr. Kaercher, EdF, described the fire protection design rules for French Nuclear Power Plants stating that they must ensure the integrity of the reactor coolant system, the capacity of safe shutdown of the reactor and the capacity of limiting offsite radioactive
release to acceptable limits. Particularly, redundant safety systems should not be affected by the fire.

After his presentation, Mr. Kaercher also described some new additions and changes in the rules concerning e.g. qualification of fire rated doors and cable protection qualification test. Operating rules will be produced covering destruction of unprotected cables and generation of random signals through cables.

Vendor

From the vendor side, Mr. Bittner, KWU, discussed the adequacy of fire protection in elder nuclear power plants. He described the development in the fire protection design of German nuclear power plants and the development of the corresponding regulatory and licensing requirements. It is clear that fire safety will be problematic for the older reactors if it is based on the requirements of actual safety standards. Mr. Bittner described a method to be used for older reactors presenting only the general and not plant specific aspects.

Research Institute

Mr. Isaksson from the Swedish National Testing and Research Institute described a way to optimize the fire protection level. He pointed out that three major steps must be considered to obtain optimal fire protection:

1. The regulations do not take into account the existence of appropriate test results which ought to be the basis for the automatic systems. Also, the manual extinguishing capacity is overestimated.
2. When guidelines refer to standards, these often are viewed as a maximum level of protection instead of a minimum standard.
3. It is extremely important that authorities possess the means and competence to evaluate the fire protection levels of nuclear power plants.

Mr. Isaksson proposed a working method using PRA/PSA to evaluate the total fire damage within each fire area and the impact on radiological safety.

T. Nilsson - Chairman of Session 2
Summary Session 3

This session contained 5 papers.

The first presentation presented by Mr. Gorza from Belgium dealt with a computer-assisted approach called "What to do in case of a fire?". This system provided much useful information concerning the correct identification of a fire, the actions to be taken, the means of safety issues and the special risk at the fire location. Also included in the system is information on inspection and maintenance of fire protection measures.

Mr. Marttila from Finland presented the improvement in fire safety at the Loviisa nuclear power plant. He talked about the weak points and the upgrading measures taken since 1990.

Mrs. Röwekamp from Germany presented the main findings from reviews of fire protection measures in older German nuclear power plants. She explained the development of the fire protection concepts in the different nuclear power plant generations. She discussed the advantages and disadvantages of automatic or manual actuation of stationary fire extinguishing systems.

Mr. Hada from Japan presented the fire protection management situation in Japanese nuclear power plants. Until now, very few fires have occurred in the utilities. Nevertheless, fire protection measures, in particular for fire prevention, retain an important place in plant safety features in Japan.

The fifth paper, presented by Mr. Magnusson from Sweden, furnished information concerning the annual report on fire protection at the Ringhals nuclear power plant. The report includes facts regarding the status of fire protection measures, fire management and co-operation with local fire authorities. Other topics of his presentation included the documentation of inspection and maintenance as well as the training of the fire brigade.

All presentations were closely related to the practice of fire protection in nuclear power plants.

H. Liemersdorf - Chairman of Session 3
Summary Session 4

Research and Development

Development of the risk analysis tools and simulation codes are being carried out to evaluate the effects caused by fire in nuclear power plants and in other industrial installations.

FREIA is an expert system developed by the Sydkraft Konsult (Sweden) to analyse the risk of accidental releases and fires in industrial buildings and power plant facilities. This expert system warns operators and provides an estimation of properties that are currently calculated.

A model filter clogging and a calculation code evaluating a confinement of a nuclear facility are being developed by the French Atomic Energy Commission. A relationship between a resistance of filter and a mass of aerosols which must be taken into account in the development of a system of fire simulation codes, was presented using constants decided by a characteristic of a combustible material and of a considered type of thermal degradation. The calculation code was developed in order to evaluate the behaviour of confinement barriers encountered in the ventilation network of a nuclear facility. It is possible to verify the risk of overpressurization, determine the thermal and mechanical stresses acting on the HEPA filters and calculate the quantities of radioactive substances released with this code.

Special effects studies during HDR fire experiments and their calculability have been performed at the Nuclear Research Centre of Karlsruhe. Much of the obtained results can be transferred to real application. It was indicated that a cable fire in the lower parts of the compartment will not attack the surroundings and a cable fire in the upper regions of cable channels will become violent when opening a door. Furthermore, it has been suggested that fire fighting would only deteriorate the situation. As a standard problem of CEC, cable fire experiments within the HDR facility are being conducted at the Technical University of Munich. The standard problem exercise has been evaluated as beneficial to demonstrate the state-of-the-art in cable fire simulation. A variety of codes has been employed in this study. Most comparisons to measurements may be considered as only partially satisfactory for the experimental phases 1 and 2. It showed that reliable models are necessary if the simulation of a cable fire involves "blind" predictions without weight loss rates and/or pyrolysis rates of involved cables.

Mr. N. Maki - Chairman of Session 4
Summary Session 5

This session contained 4 papers.

Risk Assessment

The first presentation of Mr. Taivainen from Finland dealt with the fire PSA for the nuclear power plant TVOI/II. This PSA included all safety-related equipment and their power supplies and took into consideration the relevant instrumentation systems. The analysis had shown that by upgrading fire protection measures, the risk for core damage frequency due to fire had been reduced by factor 5.

In the second presentation, Mr. Berg from Germany reported about a German approach for the application of fire probabilistic risk analyses in NPPs. These analyses make it possible to identify weak points in fire safety, e.g. critical fire zones. They provide a quantitative event tree analysis to determine the risks for core damage and melting frequencies in combination with level 1 PSA and demonstrate possible improvements to optimise fire protection.

The third presentation, by Mrs. Morales from Spain, gave an overview of the use of deterministic as well as probabilistic methods for improving fire protection in the Almaraz NPP. These analyses resulted in specific proposals for fire protection measures, in particular with regard to fire zones, accepted by the Spanish regulatory body. Thereafter, the improvements could be implemented in the Almaraz NPP for optimising fire protection and minimising the risk of core damage.

Mr. Lamuth of France presented a methodology of a fire PSA by the French Institute for Protection and Nuclear Safety (IPSN) for the French 900 MWe PWRs. The main goal of this fire PSA was to confirm and supplement the deterministic analysis and accept it as a basis for the fire protection design to identify weak points in the protection concept. As a result of those developments, a fire PSA requires knowledge of special fields such as: fire physics, probabilistics, reactor operation and the human factor.

Mrs. M. Röwekamp - GRS
Summary of the Panel Session

Mr. Hada, PWG1 Vice-Chairman from Japan, commented on the meeting from the PWG1 point of view. He pointed out the importance of everyday efforts on fire protection by the plant personnel and the fire brigades. He acknowledged the remarkable results received from research studies and experiments noting that PSA activities provide new methodology linking fire protection and nuclear safety. The remaining problems are the unsatisfactory information exchange between regulatory bodies of the member states and the insufficient information supplied relating to real fires in NPPs.

Prof. Hosser from Germany pointed out that there still exists problems concerning the fire safety of older plants. Approaches dealing with these problems vary from highly sophisticated to relatively pragmatic. As a consulting engineer on one hand, and a researcher on the other, he would try to balance these two approaches. Prof. Hosser mentioned the need for more data on fire frequencies, the behaviour of cables and fuel tanks in case of fire, and fire spreading along barriers. Although there have been improvements and much software development during the last few years, there are still many problems and unsolved weak points worldwide concerning fire safety in NPPs. He therefore proposed to work together in this field, exchange information and compare the different approaches on an international basis. It would be effective to create an international fire protection data bank containing contributions from each country.

Mr. Helleberg from Sweden underlined the significance of nuclear power safety as the main aspect for fire protection of the different plant generations. From his point of view, the main goal is reactor safety, where international guidelines (IAEA Safety Guide) exist, followed by personal safety with sufficient national regulations. A further goal is economical safety. In Sweden, four groups of fire protection items are considered: fire prevention, detection, mitigation of fire effects and manual fire fighting. For the various plant generations, it is important to reach a relatively high level of these items in order to have a safe plant. He demonstrated this by two examples.

In addition, Mr. Vandewalle from Belgium pointed out the need for further collaboration between the regulatory bodies as well as with the utilities. He made a proposal for an improved information exchange for fire protection, requesting experts to promote this cooperation in their respective countries. From his point of view, PRA/PSA should be considered as an important tool used to locate the weak points in regard to fire safety in NPPs.

Mr. Madden (USA) reported on the U.S. perspective concerning fire protection only for power operating conditions. He showed interest in the protection measures of other plant operating states e.g. cold shutdown and mid-loop operation. He proposed that PWG1 promote a communications network for various countries to solve fire protection problems on an international basis. Regarding PRA, he stated that it would be worthwhile to maintain a balance between the deterministic and probabilistic approaches.

Mr. Liemersdorf of Germany mentioned that fire protection in NPPs is a very complex issue and requires knowledge in various fields (from materials behaviour of combustibles to civil engineering and systems techniques). He noted that, according to his experience, this meeting was
the first of its kind where experts of so many different technical backgrounds came together to discuss fire safety. In agreement with Prof. Hosser and Mr. Vandewalle, he proposed a common programme to unify approaches and requirements and consequently, achieve a sufficient safety level.

Mr. Isaksson from Sweden recommended that experts in the fire safety field should meet with experts from other industries to exchange information. This idea was supported by Mr. Hada, who mentioned that few people in the nuclear industry work in this field and that the majority of advances in fire protection come from outside the nuclear industry. Mr. Müller from Germany, a member of various fire groups, underlined the importance of a fire protection data transfer between the nuclear and non-nuclear field.

Mr. Zander from Germany spoke once again about the meaning of PSA and of the problems occurring by this lack of knowledge due to the circumstances and boundary conditions (e.g. heat of combustion, velocity, etc.). Messrs. Liemersdorf and Madden agreed, mentioning as an example the separation of trains where the identification of reliabilities determines a probable uncertainty of a PRA assessment. Prof. Hosser added that PSA must be applied with caution, mainly as a relative tool, carefully comparing the results with deterministic ones taking into account the physical background. From the U.S. regulatory point of view, Mr. Madden agreed that PSA is a useful tool but the deterministic approach could not be overlooked adding that one should always try to assure that PSA is reasonable. Mr. Andersson (Sweden) pointed out that all fire incidents (e.g. a fire incident in the turbine hall of a Swedish NPP) cannot be covered by a PSA but together with a deterministic approach, PSA could be an appropriate method. Mrs. Morales of Spain agreed and mentioned that regulatory bodies frequently require a PSA, but it depends on whether that tool is a useful method. Mr. Madden added that even the U.S. NRC was requesting a PSA from the regulatory bodies. Mr. Hada stated that it would be useful to apply PSA to older NPPs.

Mr. Vandewalle added that caution should be taken with simulation models regarding fire behaviour because the circumstances influencing a real fire are not necessarily represented by the models.

Mr. Müller called for a person from a well-known international institution (IAEA, OECD) with initiative to promote fire topics. Mr. Hada recommended that voluntary experts should form a group on fire activities, i.e. holding discussions on special topics and making a first approach at the creation of a fire protection data base. From the viewpoint of Prof. Hosser, this proposal could be possible due to common interest but PWG1 would be obliged to promote it to the international nuclear community. Therefore, Mr. Hada agreed to prepare a list of fire protection requirements which could perhaps be discussed in one of the future PWG1 meetings and, of course, by the experts in each country. Dr. Kotthoff supported the proposal to present these topics to PWG1.

After a short summary of the panel discussion with conclusions by Mr. Hada, Dr. Kotthoff closed the meeting by thanking the participants for the exceptional interest shown and for the productive discussions.

Mr. M. Röwekamp - GRS
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS
PRINCIPAL WORKING GROUP N° 1 SPECIALIST MEETING

on

FIRES AND FIRE PROTECTION SYSTEMS IN NUCLEAR POWER PLANTS

Hosted by Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH

FINAL PROGRAMME AND VISIT INFORMATION

Cologne, Germany,
6th to 9th December, 1993
Specialist Meeting on Fire Protection and Fire Protection Systems in Nuclear Power Plants
Cologne, Germany, 6th to 9th December 1993

Provisional Programme

Monday, 6th December, 1993

Morning

9:00 - 9:30    Registration

9:30 - 10:00   Welcome speech given by Dr. K. Kotthoff
               NEA introduction by the Secretariat

Session #1:    General Fire Protection - Mr. A. Vandevalle, Chairman

10:15 - 10:45  "Towards Enhancing Fire Safety in Nuclear Power Plants"
               by W. Zhong

10:45 - 11:00  Coffee Break

11:00 - 11:30  "The Importance of Using Fast Response Sprinklers"
               by H. Ingason

11:30 - 12:00  "Fire Safety Study in Nuclear Power Plants in India"
               by P. K. Ghosh

12:00 - 12:30  "Water Mist Fire Extinguishing Systems"
               by M. Arvidson

12:30 - 14:00  Lunch

Afternoon

Session #2:    Design Rules and Regulatory Requirements - Mr. T. Nilsson, Chairman

14:00 - 14:30  Defense-in-Depth, A Regulatory Approach for Assuring Post-Fire Safe-Shutdown Capability by P. M. Madden

14:30 - 15:00  "Adequacy of Fire Protection in Elder Nuclear Power Plants"
               by H. Bittner and R. Wittmann
Monday, 6th December, 1993

Afternoon (Continuation)

15:00 - 15:30  "Applied Fire Protection Engineering: A Way to Optimize the Fire Protection Level" by S. Isaksson

15:30 - 15:45  Coffee Break

15:45 - 16:15  "The United Kingdom Approach to Fire Hazard Analysis" by J. S. Cowley

16:15 - 16:45  "Fire Protection Design Rules for French Nuclear Power Plants" by M. Kaercher

Tuesday, 7th December, 1993

Technical Visit
Bu transfer to NPP Philippsburg
Departure from GRS: 7:00 - Arrival at GRS: ca. 20:00

Wednesday, 8th December, 1993

Morning

Session #3:  Operating Experience and Safety Issues - Mr. Liemersdorf, Chairman

9:00 - 9:30  "What to Do in Case of a Fire?" by E. Gorza

9:30 - 10:00  "Improvements in Fire Safety at Loviisa Nuclear Power Plant" by J. Marttila

10:00 - 10:30  "Main Findings from Reviews of the Fire Protection of Older German Plants" by T. Riekert and M. Röwekamp

10:30 - 10:45  Coffee Break

10:45 - 11:15  "Fire Protection Management in Japanese Nuclear Power Plants" by M. Hada

11:15 - 11:45  "Annual Report on Fire Protection at Ringhals Nuclear Power Plant" by T. Magnusson

11:45 - 13:30  Lunch
Wednesday, 8th December, 1993

Afternoon (Continuation)

Session #4: Research and Development - Mr. Maki, Chairman

13:30 - 14:00 "FREIA - A Risk Analysis Tool for Accidental Releases and Fires in Industrial Buildings" by F. Jørud

14:00 - 14:30 "Model of Filter Clogging in Case of Fire"
by J. C. Laborde, C. Prevost and J. Vendel

14:30 - 14:45 Coffee Break

14:45 - 15:15 "Calculation Code Evaluating the Confinement of a Nuclear Facility in Case of Fires" by J. C. Laborde, C. Prevost and J. Vendel

15:15 - 16:30 Some results of CEC-Standard Problem "Predictions of Effects Caused by a Cable Fire Experiment within the HDR Facility"
by H. Karwat
"Special Effects Studies During HDR Fire Experiments and Their Calculability" (+ video) by K. Müller

Thursday, 9th December, 1993

Morning

Session #5: Risk Assessment - Mr. Zhong, Chairman

9:30 - 10:00 "Fire PSA TVO I-II" by R. Himanen, E. Nikula and K. Taivainen

10:00 - 10:30 "An Approach for the Application of Probabilistic Fire Risk Analysis in German Nuclear Power Plants" by H. P. Berg, G. Brelling and H. H. Hoffmann

10:30 - 10:45 Coffee Break

11:15 - 11:45  "Methodology of the Fire Probabilistic Safety Assessment of 900 MWe PWRs" by P. Lamuth and R. Bertrand

11:45 - 13:00  Lunch

13:00  Final Panel Session; Members: M. Hada (Chairman), H. Liemersdorf, G. Helleberg, D. Hosser, P. Madden and A. Vandevalle

14:15  Conclusions of the meeting by K. Kotthoff
Session #1

General Fire Protection

Mr. A. Vandewalle, Chairman
Towards Enhancing Fire Safety

in

Nuclear Power Plants

An introduction to the fire safety project of
the International Atomic Energy Agency

Wanli ZHONG, IAEA

FIRES AND FIRE PROTECTION SYSTEMS
IN NUCLEAR POWER PLANTS
CSNI/PWG No. 1 Specialist Meeting
OECD Nuclear Energy Agency

Cologne, Germany
6th-9th December, 1993
Towards Enhancing Fire Safety in Nuclear Power Plants

Wanli ZHONG, IAEA

Abstract

The paper introduces the newly launched fire safety project of the International Atomic Energy Agency which aims at further enhancing fire safety in nuclear power plants, especially in those which were built to earlier standards. The project is being implemented and proposed to develop guidelines for performing fire hazard analysis, as well as for conducting fire safety inspection and PSA based fire risk assessment. It further aims at developing a methodology for the systematic assessment of overall fire safety of a nuclear power plant by carrying out fire safety inspection and root cause analysis of representative fire related events occurring in the plant, and to establish a database for an exchange of information as well as for PSA based fire risk assessment.

1. INTRODUCTION

The close correlation of fire hazards and nuclear safety has been clearly recognized ever since the Browns-Ferry fire in 1975. A broad activity in all fields of fire prevention, protection and mitigation has taken place, bringing about substantial improvements. Nevertheless, the Chernobyl accident and, more recently, the fires at the Vandelllos I and Chernobyl II, the repeated occurrences at Kozloduy, as well as the fire at Narora, India once again raised questions on fire protection aspects in nuclear power plants worldwide.

Being aware of the fact that the number of fires in nuclear power plants is still large in spite of the enormous efforts made in research and development of fire safety technology as well as in fire safety upgrading of nuclear power plants in operation, and the progress made in fire safety regulations, the IAEA has recently launched a comprehensive project which attempts to tackle the issue from a different angle. The project focuses on developing a methodology for the assessment of fire safety in order to thoroughly examine the effectiveness of the overall fire protection programme of a nuclear power plant in operation, thus enabling the identification of vulnerable aspects of the programme in a systematic way.

The first part of the new project, which is already being implemented, is the development of an operation safety guide entitled "Fire Safety Inspection in Nuclear Power Plants", the purpose of which is to evaluate the overall fire protection programme of a nuclear power plant, including fire hazard analysis, passive and active fire protection measures, manual fire fighting capability, fire prevention and administrative control procedures, a quality assurance programme, as well as the assessment of implementation and effectiveness of the fire protection programme.

The second part of the project consists of developing guidelines for analyzing the fires occurring in the plant so as to identify the root causes of the representative events,
thus enabling the disclosure of deficiencies in the existing fire protection programme. The results of this analysis will cover such scopes as ergonomic and safety culture aspects not found by inspections.

The methodology for the assessment of fire safety consists essentially of a combination of the aforementioned independent but mutually supplemented guidelines.

Fire hazard analysis is the basis for the installation of active and passive fire protection measures. Nevertheless, it is not common practice for a comprehensive fire hazard analysis to be performed for a nuclear power plant built to earlier standards in both eastern and western countries. Therefore, guidelines for performing fire hazard analysis should be urgently introduced before an evaluation can take place.

Fire risk assessment based mainly on the PSA approach is also considered vital for gaining more detailed insight into the fire safety aspects of a nuclear power plant. Guidelines for a simplified PSA fire risk assessment are being proposed for inclusion in the project. In order to conduct the assessment a comprehensive database is indispensible.

The ultimate goal of enhancing fire safety of nuclear power plants worldwide is to be realized by providing services in the form of advisory missions and training courses to disseminate information on the use of the guidelines for performing analyses and inspections. A further service is to be provided by the assessment of fire safety teams whose task is to perform an assessment and offer recommendations for further improvements.

2. DEFENCE-IN-DEPTH

The general safety requirements for 1) shutting down the reactor, 2) removing residual heat, and 3) confining radioactive materials should be preserved during and after an accident. The generally recognized defence-in-depth concept is also adopted in the fire safety area and provides an overall strategy for ensuring that no fire will prevent safety systems from performing the required safety functions. To compensate for potential equipment, personnel, and procedure failures, several layers of protection against fires are provided, namely:

Layer 1. to prevent fires from starting
Layer 2. to detect and extinguish those fires which do start
Layer 3. to prevent the spread of those fires which are not extinguished and to mitigate their consequences.

The integrity of the layers is maintained by conservative design, quality assurance, safety culture, as well as an overall fire protection programme during operation. To be specific, the three layers are conserved respectively by four levels of diverse protection means as follows:

Level 1. administrative controls
Level 2. active (automatic) fire protection systems
Level 3. passive fire protection provisions
Level 4 (Backup). manual fire fighting capability.

It is understood that none of the echelons is perfect, so that if a failure should occur it would be compensated for or corrected without causing harm to individuals or the public at large in the event of a fire. Ideally, the hierarchical deployment of different echelons of equipment, personnel, and procedures should be able to prevent any fire from jeopardizing the three basic safety functions. In reality, nevertheless, fires have occurred quite frequently in nuclear power plants and some of them were safety significant and led to a loss of safety functions. This may be mainly due, inter alia, to the fact that the fire protection design of those nuclear power plants complied with earlier standards, to which the defence-in-depth principle was not properly applied. However, this can be compensated for by strengthening various echelons during operation provided that the problems have been identified.

This has stimulated the initiative of the IAEA to launch a comprehensive fire safety project to evaluate the fire safety of a nuclear power plant through a combination of systematic inspection of the implementation and effectiveness of the overall fire protection programme in place and an in-depth analysis of root causes of representative fire related events occurring in the plant. It is believed that this type of assessment will be able to identify and, in turn, to eliminate the deficiencies in the plant's fire protection programme.

It is not the matter only to blindly respect the universally true old saying that "To destroy the lion while he is yet but a whelp" but also to reflect the fact that prevention is more feasible and cost effective than measures for fire fighting and for mitigating the consequences especially for the nuclear power plants built to earlier standards. Moreover, prevention avoids any damage being caused to the nuclear power plant. The on-going Agency's fire safety project, although comprehensive, is mainly focused on preventing fires from starting.

3. THE IAEA FIRE SAFETY PROJECT

After the Browns-Ferry fire, the Agency quickened the paces to development its first Safety Guide on fire safety under the Nuclear Safety Standards (NUSS) programme, i.e. Safety Series No. 50-SG-D2 "Fire Protection in Nuclear Power Plants". The purpose of this Safety Guide is to give design and some operational guidance for protection from fire and fire-related explosions in nuclear power plants. The document was published in 1979.

The Chernobyl accident indicated the necessity that the nuclear community should be prepared for fires in high radiation fields. In fact, the former Soviet assessment of the accident constructed the main arguments for convening a symposium on "Fire Protection and Fire Fighting in Nuclear Power Installations". It was held at the headquarters of the Agency in 1989 in cooperation with several organizations working in the nuclear or fire protection fields.
With the experience of nuclear power plant operation in the past two decades and the findings of modern analysis methodologies, it is confirmed that fire related events impose a real threat to nuclear safety. It is also noticed that considerable developments in design and regulatory requirements on fire safety, and in fire protection technology and analysis techniques have taken place. The decision of revising the first edition of the 50-SG-D2 to reflect those advances was made and the 50-SG-D2 (Rev.1) came into being in 1992.

The still relatively frequent fire related events in recent years in the world show a gap between the advances in various fields of fire safety and their implementation in the nuclear power plants in operation. To narrow the gap, the Agency has embarked on a new fire safety project since 1993.

3.1 Safety Guide on "Fire Safety Inspection in Nuclear Power Plants"

It is the intention of the Agency's fire safety project to firstly provide detailed explanations and methods for evaluation of the application of the design Safety Guide 50-SG-D2(Rev.1) in a nuclear power plant in operation, and then to further extend the scope of the evaluation to cover all operational aspects. The guidelines are mainly in a checklist form. Step-wise approach is adopted to develop such comprehensive documents:

a). fire protection and fire fighting techniques
b). fire hazard analysis
c). implementation and effectiveness of fire protection programme
d). fire safety inspection in nuclear power plants

The first 3 steps will essentially cover all items need to be evaluated concerning fire safety of a nuclear power plant in operation, and the last step is to summarize and sublimated the details with the objective to form a operation Safety Guide on fire safety inspection in nuclear power plants.

The guidelines of Steps a) and b) entitled

"Guidelines for Inspection of Fire Protection and Fire Fighting Techniques for Nuclear Power Plants" and

"Guidelines for Evaluation of Fire Hazard Analysis for Nuclear Power Plants"

respectively have essentially been completed and the steps c) and d) will be implemented in 1994 and early 1995 respectively.

3.2 Guidelines on performing fire hazard analysis

The proposal of development of the guidelines is based on the fact that the older nuclear power plants in most of the countries did not have a comprehensive fire hazard analysis in place and no detailed guidance on performing the analysis is currently
available. Nevertheless, it is generally recognized that the fire hazard analysis shall be considered the basis for design of active and passive fire protection provisions as well as the ground for fire safety modification and backfitting of older plants. This proposal has received wide support from the Member States of the IAEA, and expedition of its development has been strongly recommended.

3.3 Guidelines for assessment of fire safety

The OSART (Operational Safety Review Team) and ASSET (Assessment of Safety Significant Events Team) are the major services provided by the IAEA to its Member States for enhancing operational safety and experience feedback of nuclear power plants. The compilation of the guidelines for assessment of fire safety has been proposed in order to develop a methodology, making reference to the successful experience of both ASSETs and OSARTs, for thorough examination of fire safety matters in nuclear power plants.

The guidelines for fire safety inspections will be used as a guidance by the international experts to make an objective assessment of the status of fire safety with respect to international levels of good practice. However, it is not an audit against set codes and standards, but a technical exchange of experiences and practices at the working level aimed at strengthening the programmes, procedures and practices being followed.

On the other hand, events always happen no matter how good the hardware and software provisions look like. In-depth investigation of representative fire related events to answer "why they were not prevented" will disclose their root causes, which, quite often, may not be revealed by inspections, such as deficiencies in surveillance programme, ergonomic, safety culture, and operational experience feedback.

The optimization of the two approaches, which look at fire safety aspects from different angles, will give a more complete picture of the status of the fire safety in the nuclear power plant. To eliminate the identified deficiencies, the philosophy of the methodology of assessment of fire safety is to put the emphasis on prevention wherever applicable. The measures of prevention are to strengthen administrative controls on ignition sources and combustibles including surveillance on potential degradation of insulation of I & C and electrical systems, and control the of leakage of flammable fluids. These mainly concern the improvements on programmes and procedures, as well as enhancement of safety culture of plant personnel, which are the most cost-effective and feasible means to compensate for the deficiencies of hardware barriers of defence-in-depth. However, hardware modifications will also be recommended wherever they are deemed necessary.

3.4 Guidelines for PSA based fire risk assessment

The performance of probabilistic fire risk assessment in recent years has indicated that fires contributed significantly to the core melt frequency. The identification of places
vulnerable to fire initiated events by the PSA enables to make improvements leading to greatest increase in plant fire safety. For the purpose of introducing a cost-effective and feasible means to the Member States, the development of guidelines for conducting a simplified PSA based fire risk assessment has been proposed with the objectives for identifying weaknesses in fire protection programme and prioritizing modifications to those areas which pose a threat to nuclear safety.

3.5 Database

In order to perform a simplified PSA based fire risk assessment, a basic database is necessary to be established, which includes fire initiating frequency, reliability data for all fire related equipment, fire barrier rating, etc. Some country and plant specific data may also be collected.

The incorporation of the data of fire related events into the database is also considered important. This will include the information on ignition source, material first ignited, detection and extinguishing systems, fire barriers, manual fire fighting, secondary effects, safety consequences, analysis of the causes of the event and corrective measures, etc.

The establishment of the database needs broad international cooperation and the support from all nuclear power plants and fire protection related organizations. It is believed that this internationally accessible comprehensive fire safety database will provide valuable information and benefit greatly to the nuclear community.

3.6 Services

The IAEA will provide various services aimed at enhancing fire safety of nuclear power plants at the requests of the Member States.

International and regional training courses on the better understanding and application of the various guidelines will be offered. Advisory missions on overall fire protection programme, on fire safety regulation, and on specific topics will be provided. Assessment of fire safety teams to systematically examine the fire safety of, and to recommend improvements to the host plants will be despatched upon requests.

Being aware of the importance of fires in nuclear safety matters, the Ukrainian Government requested a fire safety mission to Zaporozhe nuclear power plant in August 1993, and another request from India to host a regional training course or a workshop on specific topics has been put forward although the IAEA's new fire safety project is only about one year old. More requests are expected with the progress of the new project. To provide these services, strong support from international fire safety experts is anticipated.
*DEFENCE-IN-DEPTH STRATEGY OF FIRE SAFETY*

1st layer: prevent from starting
1ST LEVEL: ADMINISTRATIVE CONTROLS

2nd layer: detect & extinguish quickly
2ND LEVEL: ACTIVE FIRE PROTECTION SYSTEMS

3rd layer: prevent spread & mitigate consequences
3RD LEVEL: PASSIVE FIRE PROTECTION PROVISIONS

4TH LEVEL (BACKUP): MANUAL FIRE FIGHTING CAPABILITY
The Importance of Using Fast Response Sprinklers

Haukur Ingason
Swedish National Testing and Research Institute

Prepared for the
Committee on the safety of nuclear installations
Principal working group No. 1 specialist meeting on

FIRES AND FIRE PROTECTION SYSTEMS
IN NUCLEAR POWER PLANTS

Cologne, December 6 - 9, 1993

Summary

The difference between standard response sprinkler heads and quick (fast) response sprinkler heads is their response or thermal sensitivity. The quick response sprinkler head is designed to operate much faster than standard response sprinkler head and subsequently it need less water to suppress the fire. The majority of sprinkler system design today is, unfortunately, based on a "control" mode rather than a "suppression" mode. This means that the sprinkler shall not necessary suppress or extinguish the fire, only control it until a manual extinction can be made by the fire brigade on site. By using a sprinkler head equipped with a quick response heat sensitive element there is a greater change to suppress or extinguish the fire. Therefore, wherever possible we recommend to use such sprinkler heads instead of standard response sprinklers. This conclusion is simply based on the fact that it is easier to extinguish a small fire than a big fire.

The designer of a sprinkler system must not only decide on the temperature rating of sprinkler heads but as well on the responsiveness of the sprinkler. A 68 °C rated sprinkler will operate when the interior temperature of the heat sensitive element becomes 68 °C. Experiments at the Swedish National Testing and Research Institute (SP) show that the air temperature may be as high as 400 - 500 °C before the element operates on standard sprinkler heads. This means that the fire can be quite large and the water from the sprinkler may have problem to penetrate to the seat of the fire. With a quick response heat sensitive element the gas temperature is much lower when the sprinkler operates, or about 150 to 200 °C. Therefore its much easier for a quick response sprinkler to suppress a fire than a standard response sprinkler. To characterise the responsiveness of sprinklers the RTI (Response Time Index) parameter was introduced in 1980.

For nuclear power plants a sprinkler equipped with a quick response heat sensitive element may be applied wherever a deluge sprinkler system is difficult to install. This can be in turbine generator room, as a point protection of emergency power generators and emergency cooling pumps, in cable galleries or in cable spreading rooms. There exist no evidence that quick response heat sensitive elements should increase the risk for false operation or failure, on the contrary it will operate earlier and reduce the fire damage.
Classification system for sprinklers

There is a widely spread misunderstanding, even among designers and engineers, that when a fire occurs in a sprinkled building, all the sprinkler heads within the system will activate and extinguish the fire. This is not the case for automatic operating sprinkler systems as only those heads which reach the critical operation temperature (e.g. 68°C) will operate. At the same time, the gas temperature outside the sprinkler head is much higher than the operation temperature due to the thermal lag of the heat sensitive element. Usually sprinklers are provided with a heat sensitive element which consists of a glass bulb or a solder link.

During the last decade, a great deal of effort has been devoted to the subject of sprinkler thermal sensitivity. The response time index (RTI)\(^1\) is now the accepted unit of measurement and is an indicator of how quickly the sprinkler will respond to temperature changes in its surroundings. Sprinkler thermal sensitivity has long been the subject of discussion at international standards committees and final agreement of a classification system is now imminent \(^2\). The classification system proposed for sprinklers is given in Table 1.

**Table 1 Proposed classification system for sprinklers** \(^2\).

<table>
<thead>
<tr>
<th>Sprinkler Classification</th>
<th>Lower limit of RTI</th>
<th>Upper limit of RTI</th>
</tr>
</thead>
<tbody>
<tr>
<td>Quick Response</td>
<td>0</td>
<td>50</td>
</tr>
<tr>
<td>Special Response</td>
<td>50</td>
<td>80</td>
</tr>
<tr>
<td>Standard Response A</td>
<td>80</td>
<td>160</td>
</tr>
<tr>
<td>Standard Response B</td>
<td>160</td>
<td>350</td>
</tr>
</tbody>
</table>

Sprinkler thermal sensitivity is also specified and limited by a conduction factor C, defined by Heskestad and Bill \(^3\), which is a measure of heat dissipation from the sprinkler to its surroundings and in particular to the pipework. In the case of slow fire growth rates the need for the C parameter to fully describe the temperature rise in the heat sensitive element become necessary. However, in case of fast fire growth rates it is often accepted to use only the RTI parameter.

**The use of the RTI value**

The use of RTI in the modelling of a fire below a sprinkler array enables the operation of the first sprinkler to be predicted with a degree of confidence that varies with the circumstances under consideration. These circumstances can be the fire growth rate, the ceiling height, the RTI value and the distance from the fire to the sprinkler head. There are computer programmes available, which are able to calculate the operation time of the first sprinkler for different fire growth rates and geometries. However, prediction of sprinkler operations subsequent to the first, under realistic conditions, continues to be a problem to modellers.
Another application field of the RTI value is the classification of sprinklers into Quick Response, Special Response or Standard Response sprinklers in accordance to Table 1. This is accomplished by using standardised test procedures. Two different tests methods are available, The plunge test (USA) and the Rate of Rise (UK). The original test method is the plunge test and it is more commonly used test method. The sprinkler is plunged into a hot air stream in a wind tunnel and the time to operation is measured. Based on knowledge about gas temperatures, gas velocity and time to operation the RTI value is determined from equation (1).

\[ RTI = \frac{-t_r \sqrt{u}}{\ln[1 - \Delta T_e / \Delta T_g]} \]  

(1)

where \( t_r \) is the response or operation time in seconds; \( u \) is the gas (air) velocity (m/s) in the test section; \( \Delta T_e \) is the temperature rating (°C) from liquid bath test relative to ambient temperature prior to plunge and \( \Delta T_g \) is the gas temperature relative to the ambient temperature. The basic assumptions in the RTI model are that

- the heat-sensitive element is heated purely by forced convection
- all heat transferred to the heat-sensitive element is stored there and there is no conduction loss to supporting structures
- the heat-sensitive element heats isothermally, i.e. the temperature distribution within the heat-sensitive element is uniform (lumped heat capacity system)
- no additional heat is required to actuate the heat-sensitive element, such as heat of fusion for solder type sprinklers.

The model is found to work quite well for most fast growing fires whereas for slow growing fires the C parameter must be added to the model.

**Sprinkler reliability**

The rules covering the installation of sprinkler systems in for example the UK, America, Sweden or Australia are broadly similar in terms of standards or in overall intent when dealing with fires. As mentioned earlier they are for most application areas intended to control the fire and not suppress or extinguish the fire. A control mode means that the sprinkler shall not necessarily suppress the fire, but only control the fire in spread and fire size corresponding to a design area until a manual extinction can be made by the fire brigade on site. This situation can, of course, be avoided in most cases by using quick response sprinkler heads as there is a greater change to suppress the fire when the fire is small. Actually, the author can not find any benefits in waiting until the fire becomes too big in size so the sprinkler system only is capable of controlling the fire.

It is found that where differences in the standards do occur they are mainly concerned with equipment specifications, deluge rates, water storage capacities
etc. A literature survey covering the available statistical data from different countries shows that successful sprinkler performance, i.e. a control of the fire, varies from 78.2 per cent to 100 per cent. However, in most cases the statistics show that in a ordinary hazard buildings the reliability is somewhat between 95-100% and in processing industry, plants and warehouse buildings, statistics from Factory Mutual Research in USA show a reliability of about 85%. The reason for this difference is probably due to the fact that the development of storage technique, package material and building sizes have been faster than the development of the sprinkler technique. No separate statistics have been found in the literature for sprinkler performance in nuclear power plants.

Published figures on the number of heads required to control or extinguish fires in sprinkler buildings vary. Two sources claim that 60% of fires are controlled or extinguished by four heads or less and 95% by five heads or less. Despite the fact that sprinklers are designed to control the fire with a great number of sprinklers operating, the statistics show other things. The reason for this is not related to the responsiveness of the sprinklers. It is probably more related to the fact that when only 3-4 sprinkler operates water pressure is much higher than the designed water pressure and herein much higher water density. The higher pressure gives better penetration of the water through the fire plume therefore the fire is controlled within a area which is much less than the design area. However, by using the quick response sprinkler this area may be reduced further and both water damage and fire damage may be reduced.

An argument against using Quick response sprinklers instead of standard response is that there is a risk that the too many sprinklers can operate as they are much more sensitive to heat than standard sprinklers. To the best of the authors knowledge this has never been verified in experiments done so far. However, it can be interpreted from the data shown in that the influences of the RTI value on the degree of damage after a sprinkler test is considerably dependent on the RTI value. High RTI values (control of the fire) tend to give higher damages than low RTI values. Our own experience with sprinkler testing show clearly the trend that faster response gives less damage of the burning material. Therefore we recommend to use quick response sprinkler heads wherever possible in nuclear power plants. This can be in turbine generator room, as a point protection of emergency power generators and emergency cooling pumps, in cable galleries or in cable spreading rooms.

Wind shadow effects

The influence of the orientation of sprinkler frame arms relative to the direction of air flow should certainly be taken into consideration as some sprinklers might operate considerably slower in real fires than might be expected from response time tests at best positions in a wind tunnel. Due to the artificially straightened air flow in wind tunnel tests, an overestimation of the shadow effects may arise compared with the real fire situation. Heskestad and Smith state that in a real fire situation, a ceiling-mounted sprinkler would be exposed to a highly turbulent ceiling flow in which the instantaneous flow direction may vary over a wide angular range. The heat-sensitive element would be in the wake of the frame arms for only a small fraction of time during the heating interval. Consequently, the wind shadow effects, caused by the frame arms, are expected to be much less pronounced in the real fire situation than in wind tunnel tests. It is found in the wind shadow effects of the sprinkler frame arms may have considerable effects on the response time
depending on the frame orientation in the wind tunnel test situation. Wherever possible the frame should be positioned perpendicular to the most likely flow direction in a fire situation. This is of course difficult to judge but there are cases where the flow will probably be only in one direction, for example in ceilings with narrow arcade created by beams, in cable galleries or in long corridors etc.

Conclusions

Wherever possible we recommend to use sprinkler heads equipped with quick response sensitive element instead of standard response sprinkler heads. For nuclear power plants they may be applied wherever a deluge sprinkler system is difficult to install. This can be in turbine generator room, as a point protection of emergency power generators and emergency cooling pumps, in cable galleries or in cable spreading rooms.

There exist no evidence that quick response heat sensitive elements should increase the risk for false operation or failure in control or suppression of a fire, on the contrary it will operate faster and reduce the fire damage.

Wherever possible the sprinkler frame should be positioned perpendicular to the most likely flow direction in a fire situation. This can be the case in ceilings with narrow arcade created by beams, in corridors or in cable galleries.

References


Water Mist Fire Extinguishing Systems

Magnus Arvidson
Swedish National Testing and Research Institute

Prepared for the
Committee on the safety of nuclear installations
Principal working group No. 1 specialist meeting on
FIRES AND FIRE PROTECTION SYSTEMS
IN NUCLEAR POWER PLANTS
Cologne, December 6 - 9, 1993

Introduction

Water is known to be an effective Class A and B fire suppressant. This is attributable to the unique physical properties of water. However practice has shown that 100-1000 times more water are used in real-life situations than in small scale tests and theoretical models. The reason for this is that all of the water is not used to its optimum i.e. the water does not reach its boiling point and evaporate. This is mainly due to too large water droplets. The ability for the water to absorb heat is increased if the droplet diameter are lowered.

SP has been involved in testing and evaluation of water mist extinguishing systems on commission by several European manufacturers of such systems. The work, which has been concentrated on marine applications, has led to test methods dealing with accommodation areas and engine rooms (hydracarbon fires) on board ships.

Due to this work SP has developed practical knowledge of applications and limitations of water mist systems and advantages and disadvantages compared with traditional sprinkler technique. Usually 10-20 times less water are needed with water mist sprinkler systems than traditional sprinkler technology in the scenarios tested.

There is no doubt that water mist systems has a potential for many different applications. There is, however, a great need for basic research and a project has recently been started at SP in co-operation with The Institute of Fire Technology at the University of Lund to evaluate design parameters such as optimum droplet sizes, spray angles locations of nozzles and flow rates.
The physical properties of water and extinction mechanisms

The high cooling effect from water comes from its heat of vaporisation, 2260 kJ/kg, and specific heat capacity of 4,18 kJ/(kg K). Basically there are two different possibilities for a water sprays to extinguish a fire, extinction of the flame and extinction by cooling the fuel [1]. Surface cooling is mainly due to larger water droplets that are able to penetrate a fire plume and are the predominant extinction mechanisms with traditional sprinkler technology.

[1] and [2] lists the following extinction mechanisms for water mist:

- Extraction of a critical percentage of heat from the flame
- Displace of oxygen due to formation of steam
- Chemical inhibition of combustion radicals by water vapour
- Blockage of heat radiation from the flame to the burning surfaces and to unburned surfaces

Arvidson [3] noticed reduced ceiling gas temperatures with smaller droplets in a ship cabin fire scenario.

As the droplet diameter decreases the total number and the total area will increase. The droplets will evaporate and form steam faster and absorb more energy compared with larger droplets.

Table 1 through 3 shows that the total number and total area of droplets increase with smaller droplet sizes (the example is calculated for 300 g water), the time to evaporate a droplet in "smoke gas temperatures" and the velocity different sizes of droplets will reach in free fall.

<table>
<thead>
<tr>
<th>Droplet diameter</th>
<th>Total number of droplets</th>
<th>Total area</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ø=10 mm</td>
<td>570</td>
<td>0,045 m²</td>
</tr>
<tr>
<td>Ø=1 mm</td>
<td>570·10³</td>
<td>0,45 m²</td>
</tr>
<tr>
<td>Ø=0,1 mm (100 μm)</td>
<td>570·10⁶</td>
<td>4,5 m²</td>
</tr>
<tr>
<td>Ø=0,01 mm (10 μm)</td>
<td>570·10⁹</td>
<td>45 m²</td>
</tr>
</tbody>
</table>

Table 2. Time to evaporation "smoke gas temperature"

<table>
<thead>
<tr>
<th>Droplet diameter</th>
<th>Time to evaporation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ø=10 mm</td>
<td>620 s</td>
</tr>
<tr>
<td>Ø=1 mm</td>
<td>6,2 s</td>
</tr>
<tr>
<td>Ø=0,1 mm (100 μm)</td>
<td>0,062 s</td>
</tr>
<tr>
<td>Ø=0,01 mm (10 μm)</td>
<td>0,000062 s</td>
</tr>
</tbody>
</table>
Table 3. Free fall velocity

<table>
<thead>
<tr>
<th>Droplet diameter</th>
<th>Free fall velocity</th>
</tr>
</thead>
<tbody>
<tr>
<td>ø=4 mm</td>
<td>9,2 m/s</td>
</tr>
<tr>
<td>ø=1 mm</td>
<td>4 m/s</td>
</tr>
<tr>
<td>ø=0,1 mm (100 µm)</td>
<td>0,35 m/s</td>
</tr>
<tr>
<td>ø=0,01 mm (10 µm)</td>
<td>0,003 m/s</td>
</tr>
</tbody>
</table>

Holmstedt [1] has compared different extinguishing agents in a gas burner and calculated the REMP value (Required Extinguishing Media Portion, the ratio of the agent quantity to fuel quantity consumed, m_r/m_f). The lower the REMP value the more efficient the agent. A comparison between different agents is given in table 4.

Table 4. REMP values for different extinguishing agents

<table>
<thead>
<tr>
<th>Extinguishing agent</th>
<th>REMP m_r/m_f</th>
</tr>
</thead>
<tbody>
<tr>
<td>Dry powder</td>
<td>1-4</td>
</tr>
<tr>
<td>Water mist =15 µm</td>
<td>1,5</td>
</tr>
<tr>
<td>Water mist =20 µm</td>
<td>1,9</td>
</tr>
<tr>
<td>Water, large droplets</td>
<td>10-100</td>
</tr>
<tr>
<td>Halon 1301</td>
<td>4-5</td>
</tr>
<tr>
<td>Halon 1211</td>
<td>4-5</td>
</tr>
</tbody>
</table>

The value 1,5 - 2 corresponds to approximately 0,160 kg water mist/m³ of air. The corresponding value for explosion prevention is approximately to 0,320 kg water mist/m³ of air.

The results shows that water mist is a very efficient extinguishent agent per weight. But it has a poor ability to reach all parts of the flame were the burning is taking place and this seems to be the determining factor to extinguish flames. The transport and evaporation from a nozzle to the flame is a complex phenomena which depends on a lot of variables such as [1]:

- Fire induced flows
- Radiation from the flames and hot gases
- Convective heat transfer
- Absorption and scattering of radiation
- Induced flows due to mist/flame interaction (reduced pressure)
- Trust of the nozzle
Droplet sizes and its description

There are many different ways to describe droplets sizes from a nozzle which can be confusing. Some of the terms and there explanation are given in [4]:

Arithmetic Mean - is a simple weighted average based on the diameters of all the individual droplets in a spray sample.

Volume Mean Diameter - is the diameter of a droplet whose volume, if multiplied by the number of droplets, will equal the total volume of the sample.

Sauter Mean - is the diameter of a droplet whose ratio of volume to surface area is equal to that of the entire spray sample.

Volume Median Diameter - The volume median diameter is the diameters which divides the volume of the spray into two equal halves. Also known as Dv0.5.

It is important to realize that one single value, regardless of which of them that are used, is not enough to give a complete description of a water spray. The whole spray distribution is the best expression. When comparing different water sprays one should also know that the droplet distribution varies with the distance and angle from the nozzle and also that different methods of measurements can give different results.

The suggested NFPA [14] definition for a water mist system is "A fixed distribution system connected to a water supply and equipped with nozzles capable of delivering a spray where 90% of the water volume is contained in droplets with a diameter less than 400 micron".

Different ways to produce water mist

There are different ways to atomize water into small droplets, the ones common for fire suppression are:

- Hydraulic atomizing nozzles
- Pneumatic atomizing nozzles
- Impingement atomizing nozzles

The different approaches has of course advantages/disadvantages. There are different kinds of hydraulic atomizing nozzles, and generally high pressures and small orifice sizes are needed to be able to produce small water droplets. This is associated with specialized pumping equipment and distribution piping. The small orifice sizes makes these nozzles very sensitive to clogging.

Pneumatic atomizing nozzles generally produces the smallest droplets. Air or other gases (N2 has been used) are used to atomize the water. The orifice sizes are not as small as hydraulic atomizing nozzles and works with reasonable water and air pressures. However the piping installation requires both water and air pipes to each
nozzle and if large areas are intended to be covered the total air demand can be very high.

Impingement atomizing nozzles produces coarser sprays than the two other types. This type of nozzle is however very robust and the orifice size is not generally as small compared with hydraulic atomizing nozzles.

**Regulations, standards and test methods**

The development of water mist systems has highlighted the need for regulations, standards and test methods for these systems. As the first systems that were brought in to the market was aimed for use onboard ships, IMO (International Maritime Organisation) was one of the first to start the work on this. The intention of their work was that a water mist systems should be equivalent to sprinkler systems and be in compliance with the SOLAS (Safety of Life at Sea) regulation. [5] lists the following basic points as applicable to equivalent systems in accommodation areas:

- The system must be automatic in operation
- The system must be capable of both detecting the fire and of controlling or suppressing the fire
- The extinguishing media should be water and the system should be capable of using seawater
- The system should be of the wet pipe type
- The system should be capable of control or suppression under a variety of fire conditions inaccomodations areas
- The system and equipment should be designed to withstand the environment normally encountered on ships
- The system should be fed from both main and emergency sources of power
- The system should be divided into sections with stop valves, alarm functions and means for testing
- The system should be designed to cover at least 280 m²

For machinery spaces the following basic points applies:

- The system should be capable of manual release
- The system should be capable of extinguishing typical machinery space fires
- The system should be ready for immediate use and capable of continuous supply of water for the necessary time period
- The system and equipment should be designed to withstand the environment normally encountered on ships
- The system should be supplied by emergency power provided from outside of the protected space
- The system may be grouped into separate sections within a protected space with water distribution valves located at safe and readily accessible positions outside the protected space
- It should not be possible for a fire in a protected space to put the system out of action

To be able to evaluate water mist systems for the above mentioned applications, test methods have been developed for accomodation areas and machinery spaces [6] onboard ships by IMO.
NFPA, National Fire Protection Association, is an organisation that has the mission to "safeguarding people, their property and the environment from the destructive fire, using scientific and engineering techniques and education". NFPA is an independent voluntary membership, nonprofit organisation and has more than 60 000 members, mostly from US, but there are members from 70 countries [7].

The basic technical activity in NFPA is the development, publication and dissemination of timely consensus standards (there are currently more than 280) intended to minimize the possibility and effects of fire. Codes and standards are developed by technical committees and NFPA started a committee in fall 1993 to develop a standard for "Fixed Water Mist Suppression Systems". The first report from the committee is expected within two years and will contain minimum requirements for the design, installation, maintenance, and testing of water mist fixed systems.

**Tests performed with water mist systems**

During the last 2 - 3 years a lot of tests has been done throughout the world for different applications with water mist systems:

*Machinery spaces* - A numerous amount of test (approximate 300 tests) has been done by SP for different clients [8] in a mock-up simulating medium spin engines. Test has also been done by the National Fire Laboratory in Canada [2] and VTT in Finland [9].

*Gas Turbine Enclosures* - In [10] tests done by the Norwegian Fire Research Laboratory in a 70 m³ turbine hood model are presented.

*Ship cabins* - Approximate 500 tests done in a ship cabin set-up has been done by SP for different clients, tests with different types of water mist systems are presented in [3].

*Telecommunications and other electronic cabinets* - Tests has been done by Fire and Safety International [11]. Tests has been done by Fire Research Station in computer cabinets [13].

*Post-flashover compartment tests* - Tests at the Fire Technology Laboratory, VTT, in Finland has been done with different nozzles, including a water mist nozzle, against post-flashover fires in a compartment [12].

*Aircrafts* - To improve the survivability in aircrafts fire tests has been done in aircrafts [13].
References


[8] Test results not published

[9] Test results not published


OECD / NEA
CSNI / PWG1 Specialist Meeting on Fire Protection and Fire Protection Systems in NPP:s
Cologne, 6 - 9 December 1993

FIRE PREVENTION AND FIRE PROTECTION PHILOSOPHY
AND
EXCHANGE OF THE FIRE ALARM SYSTEM AT FORSMARK
UNIT 3

Olle Andersson
Göran Bergkvist

FORSMARKS KRAFTGRUPP AB
S 742 03 Östhammar
SWEDEN
FIRE PREVENTION AND FIRE PROTECTION PHILOSOPHY

General

Fire prevention and fire protection at the Forsmark Power Plant have the following goals:

Safety for the public and the environment
Prevention of accidents and by that protect the people living close to the plant from high radiation doses. This is part of the licensee responsibility.

Plant safety
Protection of all installations and measures to secure the value of the plant in the best way if a fire should occur. This is part of the owners responsibility.

Safety for the personnel
Secure the possibility for evacuation of personnel and access for fire fighters in an optimal way. This is part of the licensee responsibility.

Fire prevention and fire protection are divided in two parts, active and passive. Passive methods are separation and division into fire compartments and fire cells.

The active fire protection include fire alarm systems, automatic and manual firefighting systems, automatically actuated fire dampers and fire barriers.

Active and passive fire prevention and fire protection systems shall be combined in such a way that the goals are reached. This shall be verified by testing.

Passive fire prevention and fire protection

Fundamental for the passive fire prevention is the limitation of the fire load as well as separation and isolation of a fire if and when it should occur. The division of the plant into fire compartments and fire cells shall follow the division of the reactor safety functions. For items not important to safety the division shall be done in such a way that a fire can be isolated in the smallest possible area considering the layout of the buildings.

In order to prevent fires and to give protection from the consequences of a fire, if it should occur, the following measures shall be taken:

- The integrity of fire compartments and fire cells shall always be maintained. This imply that fire doors shall be closed and fire stops in cable and pipe entryways shall be installed and sealed in a proper way.

- The amount of combustible material in the plant shall be below the amount calculated for the specified maximal fire load. This imply a careful selection of material when plant modifications are performed and that special attention must be paid to the storage of combustible materials, such as oil, packaging material, waste etc.

It is only allowed to use combustible materials when other materials are technically or economically unsuitable.
Regular inspections shall be performed in order to verify the status of the passive fire prevention and fire protection systems. The systems shall also be included in the routine for rounds in the plant. In addition, all personnel shall inspect their own work area. Operations management shall inspect the plant periodically together with specialists from the fire brigade.

All personnel and contractors shall be informed and educated about the function and importance of the systems for passive fire prevention and fire protection.

Fire alarm systems

In order to detect a fire and provide unambiguous information to the control room personnel in the best way, the fire alarm system shall meet the following requirements:

- The fire alarm system shall cover the whole plant, i.e. all rooms where a fire can start shall be supervised by a fire alarm system common for the unit.

- The fire alarm system shall be able to detect a fire and provide relevant and correct information to the control room personnel for a design base fire (DBF).

- The fire alarm system shall provide relevant and correct information about the integrity of fire compartments and fire cells, exclusive of fire doors. Possibility to present deviations from the expected fire compartment and fire cell integrity shall be provided in such a way that the operators can take proper actions.

- The fire alarm system shall provide a possibility to disconnect fire detectors, e.g. when welding takes place. It shall be possible to perform disconnection and reconnection from the central control room. Disconnected detectors shall be substituted by manual supervision.

- Disconnection and administration of disconnected detectors shall be provided in such way that the smallest possible part of fire alarm system is affected by each disconnection. If possible the alarm system shall provide functions for the administration of disconnected detectors.

- The fire alarm system shall be able to supervise itself. Alarm shall be given if a detector is malfunctioning or if a fault is detected in other parts of the system.

- Provisions shall be made that make it possible to test the whole fire alarm system during operation. Detectors in not accessible rooms are excluded from this requirement.

In order to verify that the fire alarm system meet specified requirements functional tests shall be performed regularly and after modifications. Tests shall be based on the DBF event.

The fire alarm system shall be inspected regularly by fire fighting specialists.
Fire extinguishing systems and fire venting systems

In order to fight a fire effective and by that stop the fire from propagating to adjacent rooms and in order to keep access and escape routes free from fire gases the following requirements shall be met:

- Fire extinguishing systems shall be automatically actuated in rooms with high fire load and in cases where several safety functions are located in the same room or in rooms where systems or equipment of great value is situated. The fire extinguishing system shall be automatically actuated from the fire alarm system when this system detects a fire.

- The automatic fire extinguishing system shall have capacity to extinguish a DBF fire within a specified time.

- It shall be possible to have access to the automatic fire extinguishing system during operation and it shall be possible to test the system during operation, except extinguishing equipment located in nonaccessible rooms.

- Fire venting systems shall be automatically actuated and be able to vent specified rooms according to specification.

- It shall be possible to have access to the automatic fire venting system during operation and it shall be possible to test the system during operation, except venting equipment located in nonaccessible rooms.

- Manual fire extinguishing equipment, e.g. hand held fire extinguishers, shall be available in required number.

- The on-site and off-site fire brigade shall be available on call as stipulated. The fire brigades shall be ready to fight a fire within a specified time. Approved and verified procedures shall exist for alarming, turn-out, and fire fighting.

In order to verify the capability of the automatic fire extinguishing systems and the automatic fire venting systems, these systems shall be tested at regular intervals. Manual and automatic equipment shall also be inspected by fire fighting specialists at regular intervals. The inspection shall include a check that manual equipment is available in specified number.

Training, including necessary exercises, shall be given to the on-site and off-site fire brigade in order to ensure that they are capable to carry out their assigned tasks. In addition to training and exercises in fire fighting, cooperation exercises with the operating staff is a prioritized activity.

Fire prevention and fire protection documentation

Fire prevention and fire protection activities shall take place according to written routines. For the fire related documentation the following shall apply:
The FSAR (Final Safety Analysis Report) shall include:

- Fire analyses and descriptions.
- The safety requirements considered during design and construction as well as the safety requirements applicable during operations.
- Accident analyses.
- System description together with references to applied design standards and design criteria.
- Fire load analyses.
- Division in fire compartments and fire cells.

**Procedures**

- Fire fighting procedures on unit and site level.
- Cooperation procedures for the on-site and the off-site fire brigade.

All procedures shall be verified. The cooperation between the on-site and the off-site fire brigade shall be documented and regularly exercised.

Exercises shall be performed in order to verify the documentation. Evaluation of procedures shall be an integrated part of the overall evaluation of exercises.

**Education and training**

A prerequisite for successful manual fire fighting is knowledgeable personnel and training of critical moments.

Exercises shall be performed regularly in order to verify that everyone have the necessary knowledge. Evaluations of the exercises shall be performed and the result of these evaluations shall point out areas that need more training or changes in procedures or changes in the training program.

**Verification and testing**

The fire prevention, the fire protection and the fire extinguishing systems shall be tested at regular intervals in order to verify the capability of the systems. Manual and automatic equipment shall be inspected by fire fighting specialists at regular intervals.

Detected deficiencies shall be corrected, either by plant modifications, training, changes in procedures, changes in test procedures or combinations of all these activities.
EXCHANGE OF THE FIRE ALARM SYSTEM AT FORSMARK UNIT 3

Background

Forsmark unit 3, a 1200 MWe BWR of ABB design, went on commercial operation in August 1985. Safety functions have a capacity of 4 x 50 % and the unit is equipped with four fully redundant safety trains.

The fire prevention and fire protection is, from a nuclear safety point of view, based on physical separation, i.e. division in fire compartments and fire cells. Safety trains are fully separated and different functions on the same train are also separated and located in different fire cells. The unit consists of about 3000 rooms.

The unit was from the beginning designed to have a sophisticated computerized fire alarm system. Some 3800 detectors were installed and connected to the alarm system from the beginning and on alarm the computer system should initiate actuation of sprinklers, dampers and ventilators etc.

As a result of an agreement between the principal contractor and the customer it was decided that in connection to the commissioning of the unit a simulation of a design base fire (DBF) should be performed in order to verify the capability of the alarm system. The procedure for the simulation is described further on.

At the first test the system recognized the alarm from the first detector (out of 250 activated) and then died. No automatic functions were initiated.

After extensive modifications a new test was performed and now the system was able to present the alarm from 17 detectors out of 262 activated but almost all of the alarms were detected correctly and all of the automatic functions were performed as expected.

The evaluation of the test concluded that although the system did not meet the specification it had such a level of functionality that the safe operation of the plant could be maintained.

In parallel with the performance of the capacity tests the operations department analysed the operational aspects of the fire alarm system. This analysis resulted in a list of new functional requirements.

As a result of the partly unsuccessful capacity test and the new functional requirements a project was started with the aim to upgrade the system to an acceptable level.

Goals for upgrade of the fire alarm system

The overall requirements for the new system was:

- The system shall be able to detect a fire and provide relevant and correct information to the control room personnel for a design base fire (DBF). The system shall be able to present the alarm from all actuated detectors and all required automatic functions shall be initiated according to specification.

- The system shall provide relevant and correct information about the integrity of fire compartments and fire cells, exclusive of fire doors. Deviations from the expected fire compartment and fire cell integrity shall be presented on demand and the list shall be updated automatically.
It shall be possible to disconnect a single or a group of fire detectors from the alarm system. Disconnection and administration of disconnected detectors shall be provided in such way that the smallest possible part of fire alarm system is affected by each disconnection. The alarm system shall provide functions for the administration of disconnected detectors. It shall be possible to check the status of a detector before it is reconnected.

In order to define the requirements in a structured manner a detailed functional specification was written by the vendor and the customer together.

**Technical solution**

The original system configuration is shown in figure 1. The requirements in the functional specification resulted in the following upgrade and changes in the system configuration:

- All 10 central fire alarm units were upgraded to the latest model.

- A new supervisory computer system was installed. All alarms and all disconnections and reconnections of fire alarm detectors are handled by this computer.

  In case the supervisory system is down almost all tasks can be performed from the central fire alarm units but in a more primitive way.

- The hardware of the three original central computers (two in parallel for automatic actuation functions and one for man-machine communication) was upgraded and the software was rewritten and changed to a large extent.

The new configuration is shown in figure 2.

**Features in the new system**

In addition to fulfil the above mentioned requirements the system have the following features:

- It is possible to check detector status before a detector is reconnected.

- It is possible to administrate disconnection and reconnection of detectors by work permit identification. A detector which is disconnected by more than one work permit will not be reconnected until the last permit is turned in.

- It is possible to list all work permits irrespective of if detectors presently are disconnected or not.

- It is possible to list all faulty detectors and disconnected detectors together with administrative data.

- It is possible to sort detector listings by detector identity, by room numbers, by fire cell identifications, by fire compartment identifications, by building identifications, by work permit numbers, by detector type etc.
- It is possible to retrieve historical data for detectors and based upon that data perform preventative maintenance on detectors.
- It is possible to test all automatic functions from the central system.

Cost and time scheduling

The new system was installed and tested on schedule. The time from disconnection of the first fire alarm central to completed verification of all system functions was 22 days, see figure 3.

The total cost for the whole new system including everything was 4 447 000 Skr (500 000 USD).

The estimated cost of 3 700 000 Skr was exceeded by 20%.

The higher cost was due to added functions and depreciation of obsolete spare parts.

Verification and Validation

For different reasons it was decided to do the upgrade during operation. All work and test activities had to be carefully planned in order to minimize the operating time with a degraded fire alarm system.

A prerequisite for the project was to have a test program corresponding to the detailed functional specification.

The verification and validation program covered the following activities:
- Writing test procedures
- Factory acceptance test (1 week)
  - Functional tests to verify the specified functions.
  - Database verification to verify that the database contained correct data.
- Site tests (22 days)
  - Detector verification, performed in connection with the installation of each fire alarm central.
  - Load tests to verify that the system was capable to handle the DBF fire according to specification.
  - Other load tests to verify specified capacities.
  - Functional tests to verify the specified functions.

The time schedule for the performance of the site tests is shown in figure 3.
Simulation of a design base fire (DBF)

The DBF fire in unit 3 is assumed to be an oil fire in the turbine containment. Unit 3 have a single turbine.

Based on assumptions on how smoke will spread in the plant the DBF fire is assumed to activate about 250 fire detectors within a relatively short time.

In order to simulate the DBF fire 12 teams (2-3 persons in each team) will work in parallel with setting detectors in alarm state manually. This is performed in two ways:

1. by using "detector gas" which is sprayed into the detector.
2. by blowing cigarette smoke through pipes into the detector.

Only smoke detectors (ionizing chamber detectors) are activated in the test.

In order to get 250 detectors activated 285 detectors are scheduled for activation. Experience has shown that the teams will miss to activate 5 to 10% of their assigned detectors way the net result of activated detectors will be about 250.

The 12 teams all start at the same time (on signal from the central control room) and activate assigned detectors according to a given plan. It takes the teams about 20 minutes to set all detectors in an activated state.

The central system shall detect, give alarm, register and print out all alarms within 28 minutes and all automatic functions shall be performed without unnecessary delay. All faults and deviations from specified positions of dampers, valves and other automatically actuated equipment shall be registered and displayed to the operator on request.

During the test one team performs measurements on the central equipment and does all evaluations of data after the test. The evaluation is an elaborate task and usually takes one or two days to perform.
Fig 1. The Fire Alarm System, Before Exchange.
Fig 2 The Fire Alarm System, After Exchange.
Fig 3: Time Schedule and Operability of Fire Alarm Centrals and Automatic Functions.

Operability
Automatic Functions
(Ability to perform automatic functions as result of a fire alarm)

100%

Fr Sa Su Mo Th We Th Fr Sa Su

Verification of operability
7 days

Load tests
1 + 1 day

Functional Tests
3 days

EA-3/A
EA-3/B
EA-3/III

Operability
Fire Alarm Centrals
(Ability to detect a fire)

100%

Fr Sa Su Mo Th We Th Fr Sa Su

ca 4h ca 4h ca 4h ca 4h ca 4h
Session #2

Design Rules and Regulatory Requirements

Mr. T. Nilsson, Chairman
Enclosure 1

ABSTRACT

Title: Defense-in-Depth: A Regulatory Approach for Assuring a Post-Fire
Safe-Shutdown Capability

Author: Patrick M. Madden
Senior Fire Protection Engineer
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission

The United States Nuclear Regulatory Commission (NRC) uses a defense-in-depth
approach to assure that a reasonable level of fire protection is provided and
maintained in nuclear power plants. "Defense in depth" aims at creating a
balanced approach through adherence to the following principles:
(1) preventing fires from starting; (2) detecting fires anywhere in the plant
promptly, quickly controlling and extinguishing those fires that do occur, and
limiting the damage; and (3) designing safety systems so that even if a fire
were to spread within a plant area, the fire will not interfere with the
performance of essential plant functions. In order to ensure that the health
and safety of the public are protected from the potential consequence a fire
may have on reactor safety, each of these principles should meet certain
minimum requirements. Strengthening one of these areas can compensate in some
measure for weaknesses, known and unknown, in the other areas.

To create a balance between these principles, the NRC staff has established
fire-protection guidelines to implement operational fire-protection
programmatic aspects and incorporate fire protection features into the design
of the facility. The programmatic aspects of the fire protection program deal
with establishing operational administrative controls and procedures for
performing routine fire hazard identification inspections and controlling
transient, combustible materials and external ignition sources. In addition,
operational procedures are established to test and inspect fire-protection
features and to provide compensatory measures when these features are degraded
or inoperable. Fire-protection features will limit fire damage to equipment
used to achieve and maintain hot-and cold-shutdown conditions. Generally,
fire protection features include installing fully automatic detection;
limiting fire propagation by compartmentalizing the plant into fire areas
(e.g., fire walls, doors, dampers, penetration seals); providing fire-
resistive barrier systems (walls or raceways) to separate redundant safe-
shutdown components; installing automatic fire suppression and control systems
to limit direct fire damage to safety-related and safe-shutdown equipment and
components; and providing alternative methods of achieving safe shutdown when
redundant safe shutdown train separation cannot be separated. In addition, to
ensure prompt fire-fighting actions, plants maintain fire brigades that are
trained and equipped to respond to most routine fire emergencies.
The recent experience with raceway fire barrier deficiencies is a good example of NRC "defense-in-depth" principles providing reasonable assurance that adequate fire safety remained in place despite the degraded condition of these fire barrier systems. Currently, the NRC is taking actions to ensure that these fire-barrier systems are capable of performing their fire-resistive protective function. The insights gained through these efforts have demonstrated the need for improved qualification testing acceptance criteria; an improved understanding of the effects that these barriers may have on cable ampacity; and the understanding that variations in installation techniques may affect the overall fire resistive rating of the barrier system.
DEFENSE-IN DEPTH:
A REGULATORY APPROACH FOR ASSURING
POST FIRE SAFE SHUTDOWN CAPABILITY

by

Patrick M. Madden
Senior Fire Protection Engineer
United States Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

HISTORY OF FIRE-PROTECTION PROGRAMS AT U.S. NUCLEAR POWER PLANTS

U.S. Nuclear Regulatory Commission (NRC) regulations governing the design of nuclear power plants, consider fires and their potential impact on reactor safety. General Design Criterion 3 (GDC 3), "Fire Protection," of Appendix A to Title 10 Code of Federal Regulations (CFR) Part 50, "General Design Criteria for Nuclear Power Plants," establishes the basic performance goals to be achieved through the implementation of fire-protection programs at nuclear power plants. This criterion states, "Structures, systems, and components shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effect of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structure, systems, and components."

GDC 3 is the basis for fire safety at commercial U.S. nuclear power plants (USNPPs). This criterion governed fire protection for USNPPs until the Browns Ferry fire of March 22, 1975, which was the most serious fire to date. As a result of this fire, the NRC recognized the need to supplement GDC 3 with specific fire-protection guidance. In response to this need, the NRC developed and published in May 1976, Branch Technical Position (BTP), Auxiliary and Power Conversion Systems Branch (APCSB), 9.5-1, "Fire Protection Program." This set of fire-protection guidelines was applicable for new plants docketed after July 1, 1976. Following the development of the BTP guidelines, which identified acceptable approaches for implementing the performance criteria of GDC 3, the NRC developed and published (in August of 1976), Appendix A to BTP APCS B 9.5-1. This appendix provided specific fire-protection guidance for those plants that were docketed and operating before July 1, 1976.

The basic premise of this supplemental NRC fire-protection guidance was to establish a fire-protection program for USNPPs based on defense in depth. The defense-in-depth principle was focused on achieving an adequate balance in
(1) preventing fires from starting.

(2) detecting fires quickly, suppressing those fires that occur, extinguishing them quickly, and limiting their damage.

(3) designing safety systems so that a fire that starts despite the fire prevention program and burns for a considerable time despite the fire-protection activities will not interfere with the performance of essential plant safety functions.

In order to provide assurance that the health and safety of the public is protected from the potential consequence a fire may have on reactor safety, each of these principles must meet certain minimum requirements. Strengthening one of these areas can compensate in some measure for weaknesses, known and unknown, in the other areas.

By mid-1979, most USNPPs had implemented most of the improvements recommended by BTP APCSB 9.5-1 or Appendix A to BTP APCSB 9.5-1. However, programmatic weaknesses in implementation were recognized. To resolve these concerns, the NRC took rule-making actions and developed 10 CFR 50.48, "Fire protection," and Appendix R to 10 CFR 50, "Fire Protection for Nuclear Power Plants". Appendix R became effective on February 17, 1981, and was applicable to plants licensed to operate before January 1, 1979. For those operating plants, three specific sections of the Appendix R fire protection rule were viewed by the NRC as being so important that their implementation was deemed mandatory. These three areas of concern dealt with fire protection for safe-shutdown capability, emergency lighting, and the collection of oil from reactor coolant pumps.

DEFESE IN DEPTH — A REGULATORY APPROACH TOWARDS ASSURING POST FIRE SAFE SHUTDOWN CAPABILITY

As discussed, the defense-in-depth regulatory approach toward fire-protection is based on creating programmatic checks and balances in (1) preventing fires; (2) limiting fire damage through prompt fire detection and suppression activities; and (3) designing plant safety systems so that a fire will not prevent essential plant safety functions from being performed. The basic elements of this approach are discussed in the following sections.

FIRE PREVENTION

Fire-prevention considerations are defined operationally, through the implementation of administrative controls. Plants, as part of their overall fire-protection programs, impose administrative controls which establish the in-plant functional responsibilities for implementing fire-prevention measures. Fire-prevention programs contain the following minimal programmatic elements:
Control of Combustibles

The administrative objectives are to:

1. Prohibit bulk storage of combustible materials inside or adjacent to safety-related buildings or systems during operation or maintenance.

2. Govern the handling and limit the use of ordinary combustible materials, combustible and flammable gases and liquids, high-efficiency particulate air and charcoal filters, dry ion-exchange resins, or other combustible supplies in safety-related areas.

3. Govern the handling of and limit transient fire loads such as combustible and flammable liquids, plastic and wood products, or other combustible material in buildings containing safety-related systems or equipment during all phases of plant operations, especially during times of maintenance, modification, or refueling. For example, wood used in safety-related areas during maintenance, modification, or refueling operations should be treated with a flame retardant.

4. Immediately control the removal of all waste, debris, scrap, oil spills, or other combustible materials resulting from the work activity following completion of the work activity, or at the end of each work shift, whichever comes first. For example, equipment or supplies (such as new fuel) shipped in untreated combustible packing containers may be unpacked in safety-related areas if necessary to support an operating condition. However, all combustible material should be removed from the area immediately following unpacking.

Control of Ignition Sources

The administrative objectives are to

1. Control ignition sources by use of a flame permit system. This permit system would provide controls over welding, open flame cutting, brazing, soldering, and grinding operations. Special precautions would be required to be put in place before the start of open flame or welding operations. These special precautions would include the use of fire watches, the use of welding blankets to protect equipment or components in the area, and the isolation of combustible materials in the vicinity of the work area.

2. Control smoking areas.

3. Prohibit the use of open flames or combustion smoke for penetration seal leak testing.
Periodic Inspections

The administrative objective is to establish periodic housekeeping inspections to ensure continued compliance with the administrative controls governing the control of combustible materials in plant areas that contain safety-related equipment or equipment that is important to safety.

To satisfy this objective, fire-protection personnel on site conduct periodic inspections to (1) minimize the amount of combustible materials in safety-related areas; (2) determine the effectiveness of housekeeping practices; and (3) ensure the availability and acceptable condition of all plant fire-protection features at the plant (e.g., sprinkler systems, fire-detection systems, emergency lighting, fire-brigade equipment, fire barrier walls; fire-doors and dampers).

Fire Detection and Suppression Activities

The success of fire-damage mitigation depends on the adequacy and distribution of fire-detection devices, the distribution of plant personnel and their ability to identify and report incipient fire conditions, and the ability of plant operating personnel to take the appropriate actions to ensure that the plant is maintained in a safe condition and that fire-suppression activities are fully implemented.

Fire Detection - Overview of Capabilities

The fire-detection capability at USNPPs comprises two complementary elements: (1) the worker in the area who discovers the fire hazard or incipient stage fire and (2) fixed fire-detection system installed throughout various plant areas.

Through administrative controls, the plant establishes a general employee training program which provides limited fire-protection training to each employee on the plant site. This training instructs an individual about control actions that should be taken upon discovering a fire, including instruction on the proper method for notifying the control room. Some plants also train employees on how to use a fire extinguisher and instruct their employees on how to perform limited fire fighting.

Generally, fire/smoke-detection systems are provided for all plant areas that contain, or present a fire exposure to, safety-related or safe-shutdown equipment. These systems are required to provide an audible and visual alarm and annunciation in the control room. In order to ensure the availability of these systems should the site lose off-site electrical power, fire-detection systems have a secondary power source.

Routine surveillance inspection and testing programs ensure that these fire detection systems are operable. In the event that a system or a portion of a system fails to perform properly as a result of an operational inspection or test failure, USNPPs are required to promptly implement compensatory measures to restore a balance in defense in depth. Generally, the human fire watch is the compensatory measure. These fire watches are trained to identify fire
hazards, report fire conditions, and in some cases take incipient fire fighting actions should a fire occur.

Two types of fire watches are used at USNPPs, roving fire watches and a continuous fire watch. For impairments to fire-detection systems, a continuous fire watch is required. The continuous fire watch is posted in the plant area in which fire detection is impaired until the problem is corrected. A continuous fire watch must also be posted when a 3-hour fire barrier used to protect one train of safe-shutdown capability is inoperable and the plant area has no fire-detection capability. The continuous fire watch enhances fire prevention and can detect a fire in the area if one were to occur.

Automatic Fire Suppression Features - Overview of Capabilities

Generally, two basic types of fixed, automatic, fire-suppression systems are installed at USNPPs: water-based systems and gaseous systems. The type of automatic fire-suppression system installed in a specific area is based on the associated fire hazards and the importance the area has on shutting the reactor down safely.

Water-based systems: Automatic water-based fire-suppression systems are generally some form of automatic sprinklers (e.g., wet pipe; preaction; water spray). These systems are generally supported by a dedicated firewater supply system consisting of redundant water sources (minimum of 300,000 gallons each), redundant and diverse fire pumps (e.g., one fire pump that is electric-motor- driven and an other that is diesel engine driven), and underground supply piping. Typical plant areas that are protected by automatic sprinklers include cable spreading rooms and other areas of high cable concentration, electrical switchgear areas, turbine building areas, diesel generator areas, safety-related pump areas, and station transformers.

Automatic suppression is also used, in some cases, to separate redundant safe-shutdown functions located within the same fire area. The application of these systems will be discussed below (see Fire Safety for Plant Safety/Safe Shutdown Systems).

Gaseous Systems: The types of gaseous systems used are carbon dioxide (CO₂) and Halon. The design considerations given to these systems focus on the minimum required extinguishing agent concentrations, system agent supply and discharge distribution in the protected space, agent soak time, and ventilation control in the plant area being protected. Typical plant areas that are protected by gaseous suppression systems include cable spreading areas, diesel generator areas, switchgear areas and computer installations.

Routine inspection and surveillance testing programs ensure that these are operable. In the event that a suppression system fails to perform properly as a result of an operational inspection or test failure, USNPPs are required to promptly implement compensatory measures to restore a balance in defense in depth. Generally, a plant will stage additional manual fire-fighting equipment in the general area of the impaired fire-suppression system. In addition, a roving fire watch will be posted to supplement an operable fire detection system in the area. This roving fire watch, patrols the affected
plant area on an hourly schedule until the suppression system is fixed in order to look for fire hazards in the area of the inoperable fire-protection feature. If a fire-detection system is not operable in the area of concern, a continuous fire watch must be posted until the impairment is corrected.

**Manual Fire Fighting - Overview of Capabilities**

Fixed, manual, fire-fighting equipment is installed in all plant areas. Fire extinguishers and manual water hose stations are accessible throughout the plant. The interior, manual, hose stations are connected to the same dedicated water sources as those supplying the automatic fire-suppression systems. The individual hose stations are generally equipped with 100 feet of \( \frac{1}{2} \)-inch-diameter fire hose with a preconnected fire fighting nozzle. The fire hose is connected directly to the valve that controls water to the hose station. The fire-fighting nozzle is an electrically safe fog nozzle. Hose stations are generally reserved for firefighting use only.

Each operating USNPP is required to have a fire brigade on site. The minimum brigade size is five personnel per shift. The brigade generally comprises of a brigade leader and four other fire fighters. The brigade is routinely trained in the classroom and on the job training in interior fire-fighting techniques, proper fire-fighting techniques for plant-specific fire hazards, proper use of plant fire fighting equipment, proper use of breathing apparatus and other specialized fire-fighting equipment, fire-brigade communications, and smoke-control measures.

Annually, fire brigades at USNPPs train in an exercise that is coordinated with the local offsite fire department. In these exercises, the fire scenarios require the plant fire brigades to fully utilize their area-specific fire fighting strategies (preplans) and demonstrate command and control functions in coordination with the offsite fire department. In addition, the plant trains local fire-department personnel in the operational precautions associated with fighting fires inside the plant. This training also includes basic radiological protection training to offsite firefighters.

**FIRE SAFETY FOR PLANT SAFETY/SAFE SHUTDOWN SYSTEMS - OVERVIEW**

The purpose of this defense-in-depth element is to design plant safety systems so that a fire will not prevent essential reactor safety functions from being performed. The NRC has established basic fire-damage limits for these functions. These limits follow:

**Hot shutdown**

One train of equipment necessary to achieve hot shutdown from either the control room or emergency control station(s) must be maintained free of fire damage by a single fire.

**Cold shutdown**

Both trains of equipment necessary to achieve cold shutdown may be damaged by a single fire, but damage must be limited so that at least one train can be repaired or made operable within 72 hours.
Design-basis accident  Both trains of equipment necessary for mitigating the consequences of a design-basis accident may be damaged by a exposure fire.

These hot-and cold-shutdown damage limits are based on the need to limit damage to systems required to achieve and maintain safe shutdown from operations at 100 percent power. This need in plant design is greater that the need to limit fire damage to those systems required to mitigate the consequences of a design-basis accident.

Acceptable methods for protecting the safe-shutdown capability in operating USNPPs follow:

(1) Separation of cables and equipment and associated circuits of redundant safe-shutdown trains by a fire barrier having a 3-hour rating; or

(2) Separation of cables and equipment and associated circuits of redundant safe-shutdown trains by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazard. In addition, fire detectors and an automatic fire-suppression system must be installed in the fire area; or

(3) Enclosure of cable and equipment and associated circuits of one redundant safe-shutdown train in a fire barrier having a 1-hour rating. In addition, fire detectors and an automatic fire-suppression system must be installed in the fire area.

If these separation criteria cannot be met, USNPPs must install either alternative or dedicated shutdown capability independent of the cables, systems, or components in the area, zone, or room under consideration. This alternative or dedicated shutdown capability must be physically and electrically independent from the fire area of concern.

NRC fire protection guidance states that USNPPs should be compartmentalized into such fire areas (as fire walls, doors, dampers, and penetration seals). USNPPs, generally, use 3-hour fire barriers to separate safety-related systems from any potential fires in non-safety-related areas that could affect their ability to perform their safety function. In addition, 3-hour fire barriers are used to separate redundant trains of safety-related systems from each other so that both trains are not subject to damage from a single fire.

THE DEFENSE-IN-DEPTH PHILOSOPHY - CHECKS AND BALANCES

Fire protection at USNPPs is a non-safety-related system that is not designed to meet the single-failure criterion, defined in Appendix A to 10 CFR 50. For example, USNPPs need not install redundant sprinkler systems supplied from independent firewater systems in each area requiring this protection; redundant or two independent detection systems are not used; and two fire doors are not installed in series in a fire barrier door opening. Under fire conditions, in order to ensure that the level of fire safety provided will adequately protect the health and safety of the public and minimize the potential for a radioactive releases to the environment, diversity is achieved
through the use of active (e.g., automatic fire-suppression systems) and passive (e.g., fire-barrier walls) fire-protection features without putting total reliance on one specific feature. The fire-protection defense-in-depth philosophy ensures this diversity and compensates for the lack of redundancy. This philosophy has several levels of fire safety which compensate for programmatic weaknesses, known or unknown, and anticipates failures within these levels.

This defense-in-depth fire-safety approach toward fire protection is examined next.

**Hypothetical Cases**

The hypothetical safe-shutdown area is a corridor with safe-shutdown Train A cable trays routed along one side of the corridor and redundant Train B cable trays routed along the other side. The fire loading in this area is moderate and the major concentration of combustibles is cables. The corridor is a fire area and is separated from other plant areas by fire barriers having a 3-hour fire rating. The Train B cable trays are enclosed with a 1-hour fire barrier enclosure down the entire length of the corridor. There is areawide fire detection in the form of smoke detectors, installed along the ceiling of the corridor. Automatic fire-suppression capability, in the form of a preaction sprinkler system, is provided and manual fire-fighting equipment in the form of hose stations and fire extinguishers is available to the area.

In addition, the plant fire-prevention program is focused on controlling transient combustibles and ignition sources in safety-related and safe-shutdown areas. The program is fairly successful in reducing conditions that could contribute to or cause fires. However, this program exhibits weaknesses from time to time and this breakdown in combustible and ignition source control results in a fire in the hypothetical safe-shutdown area.

The following case studies, are arranged in order of their safety significance. In Case 1, the fire has the least impact on plant safety; the Case 6 scenario could have a significant impact on plant safety if the fire brigade cannot control the fire in a timely manner.

**Case 1—Normal Expected Fire Mitigation:** This hypothetical fire is detected in its incipient stage by the smoke detectors. The indication of fire is alarmed and announced in the main control room. The alarm is acknowledged and an operator is sent to the corridor to confirm the fire. During this time, the fire continues to grow and the temperature at the ceiling level continues to rise. The operator arrives in the area and confirms the fire to the control room via plant telephone or radio. The fire is now of sufficient size that it is affecting Train A cabling. The fire barrier enclosing Train B is protecting these shutdown cables from fire damage until the temperature at the ceiling level is great enough to actuate the sprinklers. At this time, the automatic sprinklers directly affected by the fire actuate and start applying water to control the fire and limit smoke development. The actuation of this system is alarmed and announced in the control room. Within this same time frame, the control alerts up the plant fire brigade and initiates a reactor shutdown. The fire brigade responds with the appropriate equipment (e.g.,
personnel protective equipment, breathing apparatus, additional hose lines and nozzles, smoke control equipment) needed to control and extinguish the fire. The fire brigade puts out the fire and turns off the sprinkler system.

Although there was some breakdown in defense-in-depth Principle 1 that allowed a fire to start, the diversity and interaction of the active and passive fire-protection features assured that one train of safe-shutdown capability (Train B) was maintained free of fire damage. An examination of this interaction reveals that the fire-detection equipment in the area detected the fire rapidly (defense-in-depth Principle 2). The fire barrier protecting Train B maintained it free from fire damage until the automatic sprinkler system reacted to the fire (defense-in-depth Principle 3). The sprinkler system actuated and controlled the fire (defense-in-depth Principle 2). The fire brigade responded and completes fire extinguishment (defense-in-depth Principle 2).

Case 2-Expected Fire Mitigation with Fire Detection Inoperable. Fire Watch Posted: In hypothetical Case 2, the fire detection is inoperable and a compensatory measure (i.e., continuous fire watch) is used to compensate for this weakness. In order for the assumed fire to occur, it also must be assumed that the fire watch does not recognize the fire hazard conditions until the fire has occurred. Using the same fire scenario as the one used for Case 1 (above), the fire would be detected by the fire watch, who reports the fire to the main control room via plant telephone. Since the presence of a fire is made directly by the fire watch, the control room would immediately alert the plant fire brigade. (Note: The fire watch, once notification has been made, could return to the area of concern and at his own discretion could take incipient fire-fighting actions. If the fire-fighting actions taken by the fire watch are not successful, the fire will become more severe. In addition, without the delay in confirming the fire (see Case 1), the fire brigade may arrive at the fire during its incipient stage and take corrective actions before the automatic suppression system actuates.) The fire at this point is sufficient to affect the unprotected cabling. The fire barrier enclosing Train B maintains the shutdown cables free of fire damage until the sprinklers actuate. At this time, the automatic sprinklers directly affected by the fire actuate and start applying water to control the fire and limit smoke development. The actuation of this system is alarmed and annunciated in the control room. The fire brigade responds with the appropriate equipment (e.g., personnel protective equipment, breathing apparatus, additional hose lines and nozzles, smoke control equipment) needed to control and extinguish the fire. The fire brigade puts out the fire and turns off the sprinkler system.

With the fire detection system out of service, the fire watch detected the fire and notified the control room. Posting a fire watch in the area in which fire protection is impaired is governed by plant administrative control procedures (defense-in-depth Principle 1). By compensating for this fire-protection weakness, a balance in diversity was maintained and the interaction of the active and passive fire-protection features ensured that one train of safe-shutdown capability was free of fire damage. An examination of this interaction reveals that the fire watch detected the fire and notified the
control room (defense-in-depth Principle 2). The fire barrier protecting Train B maintained it free from fire damage until the automatic sprinkler system reacted to the fire (defense-in-depth Principle 3). The sprinkler system actuated and controlled the fire (defense-in-depth Principle 2). The fire brigade responded and finished putting out the fire (defense-in-depth Principle 2).

Case 3—Expected Fire Mitigation Where Automatic Fire-Suppression Fails: With an operable fire-detection system, the hypothetical fire is detected in its incipient stage by the smoke detectors. The indication of fire is alarmed and annunciated in the main control room. The alarm is acknowledged and an operator is sent to the area of the fire alarm to confirm the fire. During this time, the fire continues to grow and the temperature at the ceiling level continues to rise. The operator arrives in the area and confirms the fire to the control room via plant telephone or radio. The fire is now of sufficient size that it is affecting Train A cabling. The fire barrier enclosing Train B is protecting these shutdown cables from fire damage. At this time, the preaction automatic sprinkler system deluge control valve fails to trip upon receiving its actuation signal. The barrier enclosing the Train B has a 1-hour fire rating and is capable of protecting the shutdown capability until the fire brigade extinguishes the fire. Within this same time frame, the control room alerts the plant fire brigade and initiates a reactor shutdown. The fire brigade responds with the appropriate equipment (e.g., personnel protective equipment, breathing apparatus, additional hose lines and nozzles, smoke control equipment) needed to control and extinguish the fire. Under these conditions, it is expected that the fire-brigade leader would ask for local offsite fire department assistance to put the fire out and that the plant would activate its emergency plan.

Under conditions where an active fire suppression system fails, diversity is tested by shifting from offensive fire-control measures to defensive measures or reliance is placed on a passive fire-resistive barrier to ensure safe shutdown capability. It should be noted that the level of fire damage associated with this fire scenario would be significantly greater than the level of damage experienced in Case 1 and 2 (above). An examination of this interaction reveals that the fire-detection equipment in the area detected the fire rapidly (defense-in-depth Principle 2). The fire barrier protecting Train B maintained it free from fire damage (defense-in-depth Principle 3). The fire brigade responded to the fire, and with the assistance of the local fire department, puts out the fire (defense-in-depth Principle 2).

Case 4—Expected Fire Mitigation With the 1-Hour Fire Barrier Degraded: In response to the hypothetical condition that has damaged or degraded the fire barrier enclosing the Train B safe shutdown cables, a roving fire watch patrols this area hourly to look for and identify fire hazards in the area of the degraded fire barrier. It is important to recognize that a fire barrier, even in a damaged or degraded state, can provide some level of fire-resistive protection. The scenario that follows assumes that the transient combustible and ignition source condition that resulted in the fire occurred 15 minutes after the fire watch had patrolled the area.
Fire is detected in its incipient stage by the smoke detectors. The indication of fire is alarmed and annunciated in the main control room. The alarm is acknowledged and an operator is sent to the area of the fire alarm to confirm the fire. During this time, the fire continues to grow and the temperature at the ceiling level continues to rise. The operator arrives in the area and confirms the fire to the control room via plant telephone or radio. The fire is now of sufficient size that it is affecting Train A cabling. The fire barrier enclosing Train B, even though it is degraded, is capable of protecting these shutdown cables from fire damage until the sprinkler system actuates. (Note: If the fire barrier was removed as part of a maintenance activity, this scenario could degrade plant safety. Total reliance would be placed on the sprinkler system to limit the level of fire damage to one train of safe-shutdown capability and maintain the other train free of fire damage.) At this time, the automatic sprinklers directly affected by the fire actuate and start applying water to control the fire and limit smoke development. The actuation of this system is alarmed and annunciated in the control room. Within this same time frame, the control room alerts the plant fire brigade and initiates a reactor shutdown. The fire brigade responds with the appropriate equipment (e.g., personnel protective equipment, breathing apparatus, additional hose lines and nozzles, smoke control equipment) needed to control and extinguish the fire. The fire brigade completes fire extinguishment and turns off the sprinkler system.

Under these conditions, the diversity of and the interaction between the active and passive fire-protection features maintained one train of safe-shutdown capability free of fire damage. An examination of this interaction reveals that the smoke detectors located in the corridor detected the fire promptly (defense-in-depth Principle 2). The fire barrier for Train B maintained it free of fire damage until the sprinklers reacted to the fire (defense-in-depth Principle 3). The sprinkler system actuated, controlling the fire (defense-in-depth Principle 2). The fire brigade put out the fire and turned off the sprinkler system (defense-in-depth Principle 2).

Case 5-Expected Fire Mitigation With Fire Detection and Automatic Fire Suppression Inoperable, Fire Watch Posted Suppression Inoperable, Fire Watch Posted: In this more severe hypothetical case, fire detection and automatic fire suppression are inoperable and a compensatory measure (i.e., continuous fire watch) is being used to compensate for these weaknesses. It is also assumed that the fire watch does not recognize the fire hazard conditions until the fire has occurred. Using the same fire scenario that was used for Case 2 (above), the fire would be detected by the fire watch and from a safe location, the fire watch reports the fire to the main control room via plant telephone. Since notification was made by the fire watch directly, the control room would immediately call up the plant fire brigade. (Note: At some USNPPs, the fire watch, once notification has been made, could return to the area on fire and at his own discretion could initiate incipient fire fighting actions. If these fire-fighting actions are not successful, the fire will become more severe.) The fire at this point is of sufficient size to be affecting the Train A cabling. The fire barrier enclosing Train B is of sufficient fire resistance to maintain the shutdown cables free of fire damage until the fire brigade controls and extinguishes the fire. Having alerted the plant fire brigade, the control room initiates a reactor shutdown. The fire
brigade responds with the appropriate equipment (e.g., personnel protective equipment, breathing apparatus, additional hose lines and nozzles, smoke control equipment) needed to control and extinguish the fire. Under these conditions, it is expected that the fire-brigade leader would ask for local offsite fire department assistance to put the fire out and that the plant would activate its emergency plan.

With the fire detection systems out of service, the fire was detected and the control room was notified by the compensatory fire watch. The posting of a fire watch in the area of a fire-protection impairment is governed by plant administrative control procedures (defense-in-depth Principle 1). By compensating for this fire-protection weakness, a balance in diversity was maintained and the interaction between the passive plant fire-protection features and the fire brigade assured that one train of safe-shutdown capability was free of fire damage. An examination of this interaction reveals that the fire watch detected the fire and notified the control room promptly (defense-in-depth Principle 2). The fire barrier for Train B maintained it free of fire damage until the fire brigade could react to the fire and place it under control (defense-in-depth Principle 3). The fire brigade responded and took action to control the fire (defense-in-depth Principle 2). The fire brigade, with assistance from the local offsite fire department, completed fire extinguishment (defense-in-depth Principle 2).

Case 6 - Expected Fire Mitigation With Fire-Detection, Automatic Fire-Suppression Inoperable and Fire Barrier Degraded. Fire Watch Posted: In this most severe hypothetical case, the fire-detection and automatic fire-suppression system in the corridor are inoperable and the fire barrier protecting Train B safe-shutdown cables is degraded. A continuous fire watch is posted in the corridor to compensate for these weaknesses. In order for the assumed fire to occur, it also must be assumed that the fire watch fails to recognize the fire hazard conditions. Using the same fire scenario is used for Case 2 (above), the fire is detected by the continuous fire watch and, from a safe location, the fire watch reports the fire to the main control room via plant telephone. Since notification was made by the fire watch directly, the control room operator would immediately alert the plant fire brigade. (Note: At some USNPPs, the fire watch, once notification has been made, could return to the area on fire and at his own discretion could initiate incipient firefighting actions. If these firefighting actions are not successful, the fire will become more severe.) The fire at this point is of sufficient size to be affecting the Train A cabling. The fire barrier enclosing Train B is of sufficient fire resistance to maintain the shutdown cables free of fire until the fire brigade controls and extinguishes the fire. Having dispatched the plant fire brigade, the control room initiates a reactor shutdown. The fire brigade responds with the appropriate equipment (e.g., personnel protective equipment, breathing apparatus, additional hose lines and nozzles, smoke control equipment) needed to control and extinguish the fires. If fire control is reasonably prompt and the integrity of the fire barrier enclosing the Train B cables is maintained, the risk to the safe shutdown capability will be averted. If the fire is not promptly controlled, risk to plant safety may be significant. Under these conditions, it is expected that the fire-brigade leader would ask for local offsite fire-department assistance to put the fire out and that the plant would activate its emergency plan.
With the fire detection systems out of service, the fire was detected and the control room was notified by the compensatory fire watch. The posting of a fire watch in the area of a fire-protection impairment is governed by plant administrative control procedures (defense-in-depth Principle 1). For this fire scenario, it was assumed that a fire occurred because the fire watch did not take any actions to eliminate uncontrolled transient combustibles and ignition sources; The worst case was assumed. Through the training provided to a fire watch, the fire watch would have recognized the potential fire hazards and would have known what actions he should have taken to correct the situation. This in itself would greatly reduced the fire risk.

This fire scenario explored the importance of maintaining a balance in defense in depth. Away from the hypothetical scene, the balance in this situation would have been maintained by the actions of the fire watch. Under the conditions identified by this case, when those actions fail, the in fire risk increases. The importance of this fire scenario and the impact it may have had on plant safety also hinges on the proficiency of the plant fire brigade. If the brigade is efficient and effective, the chances are good the fire would have been promptly controlled. Because so many weaknesses are assumed in the fire-protection defense in depth for this scenario, this fire case would have been a greater challenge to plant safety than any of the other cases discussed above.

APPLICATION OF DEFENSE-IN-DEPTH TO SPECIFIC PLANT AREAS

The basic premise of the defense-in-depth approach is to ensure that a reasonable balance in fire protection is in place that will produce an adequate level of public safety. Using the defense-in-depth principles, the NRC established levels of fire safety for specific plant areas. The sections that follow summarize recommended fire-protection features for plant specific areas and describe those features and how they meet the objectives of the defense-in-depth approach.

Cable Spreading Room

Fire Hazard and Mitigation Considerations: Electrical cables present the major fire hazard in the cable spreading rooms. These rooms are often difficult to access for manual firefighting because so many cable trays and cables are routed through the area. Fires involving cables develop significant smoke densities and generally have low to moderate heat-release rates and a slow to moderate rate of fire growth.

The fire safety defense-in-depth objectives for a USNPP cable spreading room are met in the following manner:

(1) Preventing fires from starting

The fire-prevention administrative governs program strictly controls transient combustibles and ignition sources.

(2) Detecting fire quickly, suppressing those fires that occur, extinguishing them quickly and limiting their damage.
A fire is detected promptly in a cable spreading room by means of an areawide automatic fire/smoke detection capability. These fire/smoke detection devices are alarmed locally and alarmed and annunciated in the main control room.

Inaccessibility and limited smoke control in cable spreading rooms can significantly interfere with implementation of manual fire fighting operations. Under these conditions, the fire-fighting process may not lead to successful fire extinguishment. Fixed, automatic fire-suppression systems are used to limit fire damage and fire's impact on reactor safety. Actuation of these systems is alarmed and annunciated in the main control room. Generally, in USNPP cable spreading rooms, automatic sprinklers are used to control a fire. By controlling the fire, the sprinkler system also limits the amount of smoke generated, by a cable fire.

Even though the automatic suppression system has contained the fire and limited its growth, final extinguishment is dependent on the actions of the fire brigade. Manual firefighting equipment such as portable fire extinguishers and hose stations are available in the area of cable spreading rooms.

(3) Designing safety systems so that a fire that starts and propagates despite the fire-prevention program and burns for a considerable time despite the fire-protection activities will not prevent essential plant safety functions from being performed.

Cable spreading rooms in USNPPs are important to reactor safety and safe shutdown. Typically, both redundant trains of safety and safe shutdown trains are located in the cable spreading room. In order to prevent a fire in an adjacent non-safety-related or safety-related area of the plant from affecting the safe shutdown function of the cable spreading room, 3-hour fire rated construction elements (walls, floors, ceilings) are used to separate this area from other plant areas. Doors, HVAC openings, and through penetrations in these fire barriers are also rated for 3 hours.

Alternative shutdown capability is provided for USNPP cable spreading rooms that contain both shutdown trains. This capability is physically independent from both the control room and cable spreading room. In addition, the controls for one train of safe-shutdown systems and the process monitoring instruments that are used to shutdown the reactor are electrically independent from the cable spreading room and control room. This capability essentially serves as a third train of control for operating one predetermined train of safe-shutdown equipment in the event a fire in the cable spreading room affects redundant trains.

**Electrical Switchgear Rooms**

**Fire Hazard and Mitigation Considerations:** Switchgear and electrical cables present the major fire hazards in electrical switchgear rooms. A switchgear fire can be serious and may be difficult to control until the power to the affected switchgear has been isolated. Depending on the configuration of the cable trays and their cable fill, switchgear fires can involve cables.
Fires involving cables generally develop significant smoke densities and have low to moderate heat-release rates and a slow to moderate rate of fire growth.

The fire-safety defense-in-depth objectives for a USNPP switchgear room are met in the following manner:

(1) **Preventing fires from starting**

The fire prevention administrative controls program strictly governs transient combustibles and ignition sources.

(2) **Detecting fires quickly, suppressing those fires that occur, extinguishing them quickly, and limiting their damage**

A fire is detected promptly in a switchgear rooms, by means of an areawide automatic fire/smoke detection capability. These fire/smoke detection devices are alarmed locally and alarmed and annunciated in the main control room.

The plant fire brigade has the responsibility to control and suppress a fire in this area. Manual fire fighting equipment such as portable fire extinguishers and hose stations are available in the area of switchgear rooms.

(3) **Designing safety systems so that a fire that starts and propagates despite the fire-prevention program and burns for a considerable time despite the fire-protection activities will not prevent essential plant safety functions from being performed**

Certain electrical switchgear rooms in USNPPs are important to reactor safety and safe shutdown. In order to prevent a fire in an adjacent area of the plant from affecting the safety function of safe shutdown related switchgear rooms, 3-hour fire barriers are used to separate these area from other plant areas. In addition, redundant safe shutdown related switchgear rooms are typically separated from each other by 3-hour fire barriers. Doors, HVAC openings, or through penetrations in these fire barriers are also rated for 3 hours.

**Diesel Generator Areas**

**Fire Hazard and Mitigation Considerations:** Engine fuels and lubrication systems present the major fire hazards in the diesel generator areas. Fires involving fuel and lubricating oils develop quickly with heavy smoke densities and have high heat-release rates and a high rate of fire growth.

The fire safety defense-in-depth objectives for a USNPP diesel generator room are met in the following manner:

(1) **Preventing fires from starting**

The fire-prevention administrative controls program strictly governs transient combustibles and ignition sources.
(2) **Detecting fires quickly, suppressing those fires that occur, extinguishing them quickly, and limiting their damage**

A fire is detected promptly in a diesel generator area by means of an areawide automatic fire/heat-detection capability. These fire/heat detection devices are alarmed locally and are alarmed and annunciated in the main control room.

As result of the combustible liquid hazards and the speed in which a fire could develop, automatic fire suppression systems are installed in USNPPs diesel generator areas. Generally, the diesel generator areas are protected by either preaction sprinkler systems or total flooding CO₂ systems. Actuation of these systems are alarmed and annunciated in the main control room. The function of these automatic fire suppression systems is to control a fire in the space. Final extinguishment is dependent on the actions of the fire brigade. Firefighting equipment such as portable fire extinguishers and hose stations is available in the area of diesel generator rooms.

(3) **Designing safety systems so that a fire that starts and propagates despite the fire-prevention program and burns for a considerable time despite the fire-protection activities will not prevent essential plant safety functions from being performed**

The diesel generators supply on site electrical power to safe-shutdown and safety-related equipment. To minimize the fire growth and severity in the diesel generator, special fire-protection features are provided. The fuel oil day tank for each diesel is limited to 1100 gallons. In addition, these day tanks, if they are installed in the diesel generator room, are located in a diked enclosure. This diked enclosure is generally large enough to hold the entire contents of the day tank plus an additional 10 percent to accommodate such fire-fighting agents as foam or water.

In order to prevent a fire in an adjacent area of the plant from affecting the safety function of these diesel generators, 3-hour fire barriers are used to separate these area from other plant areas. Safety is further protected by separating the redundant diesel generators from each other by 3-hour fire barriers. Doors, HVAC openings, or through penetrations in these fire barriers are also rated for 3 hours.

**Safe-Shutdown Pump Areas**

**Fire Hazard and Mitigation Considerations:**

Lubricating oils and, in some cases, heavy cable concentrations in the area of these pumps present the major fire hazards associated with safe shutdown pumps. If the fire involves the pump lubricating oils, it will develop quickly with heavy smoke densities and a high heat release and fire growth-rate. Pump fires involving lubricating oils are expected to be localized in the area of the pump. If cables in cable trays are close to the pump, where they can be influenced by the fire plume, they may also become a secondary fire source.
The fire safety defense-in-depth objectives for a USNPP safe-shutdown pump area are met in the following manner:

1) Preventing fires from starting

The fire-prevention administrative controls program strictly governs transient combustibles and ignition sources.

2) Detecting fires quickly, suppressing those fires that occur, extinguishing them quickly, and limiting their damage

A fire is detected promptly in safe-shutdown pump area by means of an areawide automatic fire-detection capability. These fire-detection devices are alarmed locally and are alarmed and annunciated in the main control room.

Depending on the plant configuration, safe shutdown pump areas may need to be protected by an automatic fire-suppression system. The fire-safety features that achieve separation between redundant safe-shutdown-related pumps are discussed further in item 3 below.

For certain safe shutdown pump configurations (i.e., redundant pumps separated by a 3-hour fire barrier), the plant fire brigade has the responsibility to control and suppress a fire. Manual firefighting equipment such as portable fire extinguishers and hose stations are available in the area of these pumps.

3) Designing safety systems so that a fire that starts and propagates despite the fire-prevention program and burns for a considerable time despite the fire-protection activities will not prevent essential plant safety functions from being performed

Certain pumps (e.g., pressurized-water reactors-charging pumps, component cooling pumps, service water pumps, auxiliary feedwater pumps; boiling water reactors-reactor core isolation cooling pump, high pressure safety injection pumps, core spray pumps, shutdown cooling pumps) are important to reactor safety and safe shutdown. Generally, to protect the safe-shutdown function of these pumps, plants have incorporated fire-safety features into the design of pump areas. USNPPs assure the availability of one train of the various pumps needed to achieve safe shutdown by separating the redundant pumps and associated cables from each other by providing (1) fire barriers having a 3-hour rating or (2) a horizontal distance of more than 20 feet with no intervening combustibles or fire hazard. In addition to the spatial separation, the area of concern is covered by area fire detectors and protected by an automatic fire-suppression system; or (3) a 1-hour fire barrier and enclosing associated cabling of one pump train in a 1-hour cable tray or raceway fire barrier system. In addition to the fire barriers, the pump area has area fire detectors and an automatic fire-suppression system.

When fire-barrier wall, floor, or ceiling systems are used to separate safe shutdown functions, all doors, HVAC openings, and through penetrations in the fire barriers are also fire rated. The fire rating of the devices protecting openings in fire barriers are at least equivalent to the fire rating of the fire barrier.
RESTORING DEFENSE-IN-DEPTH — RESOLVING THE WEAKNESSES IN CABLE TRAY AND RACEWAY FIRE BARRIER SYSTEMS

Background — Discovery of the Issue

As a result of industry events involving Thermo-Lag 330-1 fire-barrier systems, the NRC established a special review team to further review and evaluate technical issues related to these fire-barrier system. The NRC's review identified the following technical issues associated with Thermo-Lag 330-1 fire barriers: (1) incomplete or indeterminate fire-test results, (2) questionable ampacity derating test results and a wide range of documented ampacity derating factors, (3) some barrier installations that were not constructed in accordance with vendor-recommended installation procedures, (4) incomplete installation procedures, and (5) as-built fire-barrier configurations that may not have been qualified by valid fire-endurance tests or evaluated in accordance with NRC guidance.

Subsequently, Texas Utilities Electric Company (TU Electric) conducted a series of full-scale fire-endurance tests to qualify the Thermo-Lag 330-1 electrical raceway fire-barrier configurations installed at its Comanche Peak Steam Electric Station. The NRC also conducted a series of small-scale fire tests of 1-hour and 3-hour Thermo-Lag prefabricated panels at the National Institute of Standards and Technology to assess the fire performance of the panels. The results of these fire tests raised additional concerns about the ability of Thermo-Lag 330-1 fire barriers to provide fire protection according to their specified fire-resistance ratings.

The NRC issued the results of the TU Electric and the NRC fire tests to USNPP licensees and identified apparent failures of Thermo-Lag 330-1 fire barriers and materials during fire-endurance testing. The NRC requested that each plant licensee using 1-hour or 3-hour prefabricated Thermo-Lag 330-1 panels or conduit shapes for raceway, wall, ceiling, or equipment enclosure fire barriers implement, in accordance with plant procedures, compensatory measures (fire watches) until the fire barriers can be qualified and declared operable.

Plan to Resolve the Fire — Barrier Technical Issues

The NRC has developed an action plan to resolve the technical issues associated with cable tray and raceway fire-barrier systems used to separate redundant safe-shutdown functions within the same fire area. This action plan, as it deals with the technical issues discussed above, consists of three parts as follows:

- Part I-Review and evaluate technical issues.
- Part II-Conduct NRC sponsored small-scale fire tests.
- Part III-Prepare inspection guidance to assist the NRC field offices.
Part I—Technical Issues Resolution:

The objective of Part I is to coordinate resolution of the technical issues with the industry, to monitor and review industry's actions, and to ensure that these actions adequately resolve the technical issues associated with fire barrier systems used to separate redundant safe-shutdown functions within the same fire area. The major focus of this work is associated with resolving the technical issues associated with the Thermo-Lag material.

Fire-Endurance Test Acceptance Criteria

One of the major tasks under Part I of the action plan is to clarify the NRC position on fire-endurance testing acceptance criteria for fire-barrier systems used to separate redundant safe-shutdown functions within the same fire area. The NRC staff is working on this and the position is scheduled to take its final form by December 1993.

This NRC position will (1) clarify the applicability of the test acceptance criteria given in Generic Letter 86-10 to raceway fire barrier systems, (2) specify a set of fire-endurance test acceptance criteria which are acceptable for demonstrating that fire-barrier systems can perform their required fire-resistive function and maintain the protected safe-shutdown train free of fire damage, (3) specify acceptable options for hose-stream testing, and (4) specify acceptable criteria for functionality testing of cables when a deviation would be necessary, such as if the fire-barrier temperature-rise criteria are exceeded and the cable sustains visible thermal damage.

The test methods and acceptance criteria specified are acceptable for determining the adequacy of fire-barrier systems proposed by USNPPs in the future to satisfy NRC fire-protection rules and regulation.

The fire-endurance qualification test is successful if the following conditions are satisfied:

1. The internal temperature of the fire-barrier system, as measured on the exterior surface of the raceway or component, did not rise more than 139 °C [250 °F] above its initial temperature; and

2. A visual inspection of the protected component or cables revealed no signs of degraded conditions from the thermal effects of the fire exposure; and

---

1 The 163 °C [325 °F] temperature condition specified in GL 86-10 was established by allowing the temperature on the unexposed side of the barrier to rise 139 °C [250 °F] above ambient laboratory air temperature which was assumed to be 24 °C [75 °F] during the fire test.

2 Examples of thermal degradation of cable jacket and insulation materials are: swollen, split, cracked, blistered, melted, or discolored jacket; exposed shield; exposed, degraded, or discolored conductor insulation; and exposed copper conductor.
(3) The fire-barrier system remained intact during the fire-exposure and hose-stream tests without developing any openings through which the protected component, raceway, or cables are visible.

For raceway fire-barrier systems, the staff adopted the hose-stream testing methodology specified in NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 9.5.1, "Guidelines for Fire Protection for Nuclear Power Plants," Revision 2, July 1981, Position 5.a. This SRP position established the acceptability of using the fog-nozzle method for hose-stream testing of fire-barrier penetration seals. The fog-nozzle hose-stream method is an acceptable option for tests of the entire raceway fire-barrier system under the proposed NRC position.

This position also clarifies that, if cables show signs of thermal degradation during the fire test, USNPP licensees can submit to the staff for review a deviation based on a demonstration of the functionality of the thermally degraded cables. This position provides specific guidance for demonstrating cable functionality, including subjecting the cables to meger and high-potential tests. The results of these tests can be used to determine the insulation-resistance characteristics of the thermally damaged cable and to determine if the cable insulation would have been sufficient to maintain circuit functionality at rated voltage during and after the fire exposure.

Nuclear Management Resources Council (NUMARC) Program

In an effort to resolve the technical issues, NUMARC is coordinating the industry's program. This program consists of performing 1-hour and 3-hour fire endurance tests on representative in-plant configurations, assessing ampacity derating, and assessing of the combustibility of Thermo-Lag. The initial focus of this program was determining, on a generic perspective, how these fire barriers were constructed and installed in USNPPs. In addition, the industry determined the range of sizes and types of cable trays and raceways protected, and representative configurations to be included in the test program.

The NUMARC test program consists of two phases. Phase 1 testing is focused on developing fire-barrier upgrades for existing Thermo-Lag installations. These upgrades consist of either applying an additional layer of Thermo-Lag fire-barrier material over the baseline Thermo-Lag fire-barrier system, stress-skin upgrades on joints and seams, and stitching of joints and seams. The Phase 1 program, completed in October 1993, tested four 3-hour and three 1-hour specimens. Preliminary results indicate that certain configurations exceeded the unexposed-side temperature criterion. NRC has not completed its final review of these tests.

Currently, NUMARC is planning its Phase 2 program. It is anticipated that the Phase 2 program will test 1-hour and 3-hour Thermo-Lag baseline fire-barrier configurations to determine the actual fire-resistive rating of these assemblies. In addition, during this phase, NUMARC will test Thermo-Lag baseline configurations upgraded with fire-barrier materials other than Thermo-Lag. Ten tests are scheduled for Phase 2.
Part II—NRC Testing

The objective of Part II was focused on assessing the combustibility and the fire-resistive performance of 1-hour and 3-hour Thermo-Lag 330-1 fire-barrier material by subjecting these materials to combustibility tests and small-scale fire-endurance tests.

In order to gain a basic understanding of the fire-resistive performance of 1-hour and 3-hour Thermo-Lag 330-1 pre-shape fire-barrier panels, the NRC, in cooperation with the National Institute of Standards and Technology, conducted a series of small-scale ASTM E-119 fire-endurance tests. The scope of these tests focused on the fire-resistive properties of the fire-barrier material. The results of these tests indicated that this material did not meet the unexposed-side temperature acceptance criterion of GL 86-10.

The NRC fire-protection requirements for certain fire-barrier installations require the use of non-combustible fire-barrier materials. To determine if Thermo-Lag was combustible, the NRC tested its combustibility. These tests were conducted in accordance with ASTM E-136, "Behavior of Materials in a Vertical Tube Furnace at 750°F." This test is a pass/fail test. For the test specimen to be considered non-combustible, the specimen must meet the following criteria: (1) The recorded temperatures of the surface and interior thermocouples do not at any time during the test rise more that 54°F (30°C) above the furnace temperature at the beginning of the test; (2) there is no flaming from the test specimen after the first 30 seconds, and (3) when the weight loss of the test specimen during testing exceeds 50 percent, the recorded temperatures of the surface and interior thermocouples do not at any time during the test rise above the furnace air temperature at the beginning of the test, and the specimen does not flame. The Thermo-Lag fire-barrier material did not pass these criteria.

Part III—Inspection Program

The objectives of Part III to develop fire-barrier inspection guidance to assist NRC regional field offices in their inspection and evaluation of the adequacy of fire-barrier configurations installed in USNPPs.

The NRC is currently developing guidance for inspecting of fire-barrier systems used to separate redundant trains of safe-shutdown systems within the same fire area. This inspection guidance is focused on assessing the adequacy of installed fire-barrier systems to perform their specified fire resistive function and their ability to protect the required safe-shutdown function in the event of a plant fire.

This inspection effort will require the following: review and evaluation of plant-related fire endurance and ampacity derating qualification test reports to determine the validity of the test reports and the applicability of the test results to the plant design; verification of the fire-barrier system installations through the review of installation procedures and specifications and quality control and quality assurance records; inspection and walkdown of installed fire-barrier systems; evaluation of the seismic analysis performed for these fire-barrier systems; evaluation of ampacity derating affects on
cables protected by these fire-barrier systems; and an evaluation of plant specific engineering evaluations which were performed to justify in-plant fire-barrier installation conditions that vary from the tested configurations.

The NRC actions are still ongoing. Current plans indicate that this action plan will be completed by the end of the second quarter of 1997.

CONCLUSION

The defense-in-depth approach assures that there is sufficient fire-safety diversity in design and operation of the plant. This diversity can adequately compensate for programmatic weaknesses, inoperable fire-protection features, and limited unexpected fire-protection failures during actual fires. The defense-in-depth fire-protection approach creates an interaction between programmatic administrative controls, fire-detection and suppression capabilities, and passive plant fire-protection features which effectively reduces the fire risk and offers reasonable assurance that one train of systems needed to achieve and maintain safe-shutdown conditions will be free of fire damage.
REFERENCES

Nuclear Regulatory Commission

May 1, 1976
Branch Technical Position (APCSB) 9.5-1, "Fire Protection Program."

February 24, 1977
Appendix A to Branch Technical Position APCS 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976."

February 19, 1981
10 CFR 50.48, "Fire protection."

February 19, 1981
Appendix R to 10 CFR 50, "Fire Protection for Nuclear Power Plants."

July 1981
NUREG-0800, Standard Review Plan (SRP), 9.5.1, "Fire Protection for Nuclear Power Plants."

April 24, 1986
"Implementation of Fire Protection Requirements" (Generic Letter 86-10).

American Society for Testing and Materials (ASTM)


American Nuclear Insurers (ANI)

ANI Information Bulletin No. 5 (79) test criteria for "Fire Endurance Protective Envelope Systems for Class 1E Electrical Circuits."

National Fire Protection Association (NFPA)

DEFENSE-IN-DEPTH:
A REGULATORY APPROACH FOR ASSURING POST FIRE SAFE SHUTDOWN CAPABILITY

BY
PATRICK M. MADDEN
SENIOR FIRE PROTECTION ENGINEER
UNITED STATES NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
OVERVIEW

- HISTORY OF U.S. FIRE PROTECTION REGULATIONS
- DEFENSE-IN-DEPTH AND ITS FIRE PROTECTION ELEMENTS
- DEFENSE-IN DEPTH PHILOSOPHY - CHECKS AND BALANCES
- CURRENT FIRE BARRIER TECHNICAL ISSUES AND PLANS FOR RESOLUTION
- CONCLUSION
HISTORY OF U.S. FIRE PROTECTION REQUIREMENTS

PRE - BROWNS FERRY FIRE

• APPENDIX A TO 10 CFR 50, GENERAL DESIGN CRITERION 3, "FIRE PROTECTION"

  - STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED TO MINIMIZE THE PROBABILITY OF FIRE AND EXPLOSION
  - NONCOMBUSTIBLE AND HEAT RESISTANT MATERIALS SHALL BE USED
  - FIRE DETECTION AND FIRE FIGHTING SYSTEMS SHALL BE PROVIDED TO MINIMIZE THE ADVERSE EFFECT OF FIRE ON PLANT SAFETY
HISTORY - CONTINUED

POST - BROWNS FERRY FIRE (MARCH 22, 1975)

- 1976 BTP AUXILIARY AND POWER CONVERSION SYSTEMS BRANCH (APCSB) 9.5-1, "GUIDELINES FOR FIRE PROTECTION FOR NUCLEAR POWER PLANTS" APPENDIX A TO BTP APCSB 9.5-1

- 1980 10CFR 50.48 AND APPENDIX R

- 1981 BTP CMEB 9.5.1 (STANDARD REVIEW PLAN), "GUIDELINES FOR FIRE PROTECTION FOR NUCLEAR POWER PLANTS"

- 1986 GENERIC LETTER 86-10, "IMPLEMENTATION OF FIRE PROTECTION REQUIREMENTS"

DEFENSE-IN-DEPTH USED AS BASIS FOR DEVELOPING FIRE PROTECTION PROGRAM GUIDANCE AND REQUIREMENTS AFTER THE BROWNS FERRY FIRE.
DEFENSE-IN-DEPTH FIRE PROTECTION PRINCIPLES

1. PREVENTION OF FIRES

2. LIMITING FIRE DAMAGE THROUGH PROMPT FIRE DETECTION AND SUPPRESSION

3. DESIGNING PLANT SAFETY SYSTEMS SO FIRE WILL NOT PREVENT SAFETY FUNCTION FROM BEING PERFORMED.

STRENGTHENING ONE OF THESE AREAS CAN COMPENSATE IN SOME MEASURE FOR WEAKNESSES, KNOWN AND UNKNOWN, IN THE OTHER AREAS.
PREVENTION OF FIRES

ELEMENTS ....

CONTROL OF COMBUSTIBLES

CONTROL OF IGNITION SOURCES

PERIODIC FIRE INSPECTIONS

USE OF FIRE WATCHES
FIRE DETECTION AND SUPPRESSION

ELEMENTS ....

DETECTION

- ADEQUATE DISTRIBUTION OF FIRE DETECTION DEVICES

- PLANT PERSONNEL (ADEQUATE TRAINING ON HOW TO REPORT A FIRE AND WHAT ACTIONS TO MITIGATE ITS CONSEQUENCE)

- USE OF CONTINUOUS FIRE WATCHES WHEN AUTO FIRE DETECTION IS INOPERABLE OR NOT PROVIDED

SUPPRESSION

- USE OF AUTOMATIC FIRE SUPPRESSION SYSTEMS (HIGH FIRE HAZARD AREAS AND AREAS IMPORTANT TO PLANT SAFETY)

- MANUAL FIRE FIGHTING CAPABILITY (HOSE STATIONS AND PORTABLE FIRE EXTINGUISHERS)

- TRAINED ON-SITE FIRE BRIGADE
DESIGNING PLANT SAFETY SYSTEMS

ELEMENT ....

FIRE SEPARATION BETWEEN REDUNDANT SAFE
SHUTDOWN FUNCTIONS (OPTIONS)

- EACH REDUNDANT FUNCTION IN SEPARATE 3-HOUR FIRE AREA, OR

- REDUNDANT FUNCTIONS WITHIN SAME FIRE AREA, ONE IS ENCLOSED IN
  A 3-HOUR FIRE BARRIER, OR

- REDUNDANT FUNCTIONS WITHIN SAME FIRE AREA, ONE IS ENCLOSED IN
  A 1-HOUR FIRE BARRIER AND AREA IS PROTECTED BY AUTO DETECTION
  AND SUPPRESSION, OR

- REDUNDANT FUNCTIONS WITHIN THE SAME FIRE AREA HAVE OVER 20
  FEET HORIZONTAL SEPARATION WITH NO COMBUSTIBLES INTERVENING
  AND AREA IS PROTECTED BY AUTOMATIC DETECTION AND
  SUPPRESSION.
DEFENSE-IN-DEPTH PHILOSOPHY
CHECKS AND BALANCES

HYPOTHETICAL CASES

CORRIDOR - ONE TRAIN OF SAFE SHUTDOWN CAPABILITY PROTECTED BY A 1-HOUR FIRE BARRIER, AREA PROTECTED BY AUTO DETECTION AND SUPPRESSION

(CASE 1 - NORMAL EXPECTED FIRE MITIGATION)

- FIRE DETECTED BY DETECTION SYSTEM.
- FIRE BARRIER PROVIDES FIRE RESISTIVE (PASSIVE) PROTECTION UNTIL AUTO SUPPRESSION ACTUATES AND CONTROLS THE FIRE.
- FIRE BRIGADE Responds AND EXTINGUISHES THE FIRE.

(CASE 2 - FIRE MITIGATION WITH FIRE DETECTION INOPERABLE, CONTINUOUS FIRE WATCH POSTED IN THE AREA)

- FIRE DETECTED BY THE FIRE WATCH AND REPORTED TO CONTROL ROOM.
- FIRE BARRIER PROVIDES FIRE RESISTIVE (PASSIVE) PROTECTION UNTIL AUTO SUPPRESSION ACTUATES AND CONTROLS THE FIRE.
- FIRE BRIGADE Responds AND EXTINGUISHES THE FIRE.
HYPOTHETICAL CASES - CONTINUED

CASE 3 - FIRE MITIGATION WITH FIRE DETECTION AND AUTOMATIC SUPPRESSION INOPERABLE, CONTINUOUS FIRE WATCH IS POSTED

FIRE DETECTED BY THE FIRE WATCH AND REPORTED TO CONTROL ROOM.

CONTROL ROOM PROMPTLY ALERTS ON-SITE FIRE BRIGADE.

FIRE BRIGADE PROVIDES FIRE-RESISTIVE (PASSIVE) PROTECTION UNTIL OFF-SITE FIRE DEPARTMENT ASSISTANCE REQUESTED TO ASSIST WITH FINAL FIRE EXTINGUISHMENT.

FIRE CONTROL AND EXTINGUISHMENT CAN BE ACCOMPLISHED BY THE FIRE BRIGADE.

OFF-SITE FIRE DEPARTMENT ASSISTANCE REQUESTED TO ASSIST WITH FINAL FIRE EXTINGUISHMENT.
INSIGHTS

• MAINTAINING A BALANCE IN DEFENSE-IN-DEPTH IS IMPORTANT FACTOR TO PLANT SAFETY AND FIRE RISK REDUCTION.
  - MOST FIRE PRAs ARE BASED ON THE ADEQUACY OF THE FIRE PROTECTION (DEFENSE-IN-DEPTH) PROGRAM (e.g., FIVE METHOD ASSUMES DEFENSE-IN-DEPTH ELEMENTS ARE IN PLACE)

• FIRE WATCHES ARE IMPORTANT TO ASSURE PROMPT DETECTION OF A FIRE (WHEN FIRE DETECTION IS INOPERABLE)

• PASSIVE FIRE BARRIERS ARE RISK IMPORTANT WHEN FIRE SUPPRESSION IS EITHER DELAYED OR INEFFECTIVE.

• ON-SITE FIRE BRIGADES ARE CONSIDERED ESSENTIAL ELEMENT IN FIRE CONTROL AND EXTINGUISHMENT.
RESOLVING THE WEAKNESSES IN CABLE TRAY AND RACEWAY FIRE BARRIER SYSTEMS

TECHNICAL ISSUES

- CAPABILITY OF THESE FIRE BARRIERS TO PERFORM THEIR FIRE RESISTIVE FUNCTION (MAY NOT HAVE BEEN QUALIFIED BY FIRE TEST)
- AMPACITY DERATING
- INADEQUATE INSTALLATION PRACTICES

NRC ACTION PLAN

- PART I RESOLUTION OF TECHNICAL PROBLEMS
  - COORDINATE RESOLUTION OF TECHNICAL ISSUES WITH INDUSTRY
  - CLARIFY AND DEVELOP A NRC POSITION ON FIRE ENDURANCE TEST ACCEPTANCE CRITERIA

- PART II CONDUCT NRC SPONSORED SMALL-SCALE FIRE TEST PROGRAM (TEST OF THERMO-LAG PANELS TO ASTM E-119)

- PART III PREPARE INSPECTION GUIDANCE FOR NRC FIELD OFFICES
FIRE BARRIERS USED TO SEPARATE SAFE SHUTDOWN FUNCTIONS WITHIN THE SAME FIRE AREA

SAFETY OBJECTIVE

ENSURE THAT A SAFE SHUTDOWN TRAIN IS FREE OF FIRE DAMAGE BY PROVIDING EITHER A 1-HOUR OR 3-HOUR FIRE BARRIER

FIRE BARRIER DESIGN OBJECTIVES

WHEN SUBJECTED TO A STANDARD TEST FIRE (ASTM E-119 OR NFPA 251), THE FIRE BARRIER SHOULD LIMIT THE HEAT TRANSFER FROM THE FIRE BARRIER MATERIAL'S EXPOSED SURFACE TO ITS UNEXPOSED SURFACE.

MAINTAIN ITS INTEGRITY DURING THE FIRE EXPOSURE AND HOSE STREAM TEST
RACEWAY FIRE BARRIER ACCEPTANCE CRITERIA

PROPOSED STAFF POSITION

• NFPA 251 (ACCEPTANCE CRITERIA USED AS BASIS)

• CRITERIA

LIMIT UNEXPOSED FIRE BARRIER SURFACE TEMPERATURE TO LESS THAN 250 °F RISE

FIRE BARRIER SYSTEM REMAINED INTACT DURING THE FIRE EXPOSURE AND WATER HOSE STREAM TEST WITHOUT DEVELOPING ANY OPENINGS THROUGH WHICH THE RACEWAY OR COMPONENT (e.g. CABLE) IS VISIBLE
RACEWAY FIRE BARRIER ACCEPTANCE CRITERIA

PROPOSED STAFF POSITION

- PROVIDES GUIDANCE FOR DETERMINING CABLE FUNCTIONALITY

- METHODS WHEN CABLES ARE NOT INCLUDED IN RACEWAY TEST SPECIMEN
  - COMPARISON OF INTERNAL TEMPERATURE PROFILES TO EQ AND LOCA TEST RESULTS
  - PERFORMING AIR OVEN TESTS OF PLANT SPECIFIC CABLES AT RATED VOLTAGE WITH MEGGER AND HI-POT TESTS

- METHOD WHEN CABLES ARE INCLUDED IN RACEWAY TEST SPECIMEN
  - CABLE INSULATION TESTING (MEGGER AND HI-POT CABLE TESTING)

BEFORE AND AFTER THE FIRE EXPOSURE FOR POWER AND CONTROL CABLES

BEFORE, DURING AND AFTER THE FIRE EXPOSURE FOR INSTRUMENTATION CABLES
CONCLUSIONS

- DEFENSE-IN-DEPTH APPROACH ASSURES THAT THERE IS SUFFICIENT FIRE-SAFETY DIVERSITY IN DESIGN AND OPERATION OF THE PLANT.

- THIS DIVERSITY CAN COMPENSATE FOR PROGRAMMATIC WEAKNESSES, INOPERABLE FIRE PROTECTION FEATURES, AND LIMITED FIRE PROTECTION FAILURES DURING ACTUAL FIRE.

- CREATES A INTERACTION BETWEEN FIRE PREVENTION, FIRE DETECTION AND SUPPRESSION AND PASSIVE FIRE PROTECTION THAT EFFECTIVELY REDUCES FIRE RISK.
DEFENSE-IN-DEPTH:

A REGULATORY APPROACH FOR ASSURING
POST FIRE SAFE SHUTDOWN CAPABILITY

BY

PATRICK M. MADDEN

SENIOR FIRE PROTECTION ENGINEER
UNITED STATES NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
OVERVIEW

- HISTORY OF U.S. FIRE PROTECTION REGULATIONS
- DEFENSE-IN-DEPTH AND ITS FIRE PROTECTION ELEMENTS
- DEFENSE-IN DEPTH PHILOSOPHY - CHECKS AND BALANCES
- CURRENT FIRE BARRIER TECHNICAL ISSUES AND PLANS FOR RESOLUTION
- CONCLUSION
HISTORY OF U.S. FIRE PROTECTION REQUIREMENTS

PRE - BROWNS FERRY FIRE

- APPENDIX A TO 10 CFR 50, GENERAL DESIGN CRITERION 3, "FIRE PROTECTION"

- STRUCTURES, SYSTEMS, AND COMPONENTS DESIGNED TO MINIMIZE THE PROBABILITY OF FIRE AND EXPLOSION

- NONCOMBUSTIBLE AND HEAT RESISTANT MATERIALS SHALL BE USED

- FIRE DETECTION AND FIRE FIGHTING SYSTEMS SHALL BE PROVIDED TO MINIMIZE THE ADVERSE EFFECT OF FIRE ON PLANT SAFETY
HISTORY - CONTINUED

POST - BROWNS FERRY FIRE (MARCH 22, 1975)

- 1976  BTP AUXILIARY AND POWER CONVERSION SYSTEMS BRANCH (APCSB) 9.5-1, "GUIDELINES FOR FIRE PROTECTION FOR NUCLEAR POWER PLANTS" APPENDIX A TO BTP APCSB 9.5-1
- 1980  10CFR 50.48 AND APPENDIX R
- 1981  BTP CMEB 9.5.1 (STANDARD REVIEW PLAN), "GUIDELINES FOR FIRE PROTECTION FOR NUCLEAR POWER PLANTS"
- 1986  GENERIC LETTER 86-10, "IMPLEMENTATION OF FIRE PROTECTION REQUIREMENTS"

DEFENSE-IN-DEPTH USED AS BASIS FOR DEVELOPING FIRE PROTECTION PROGRAM GUIDANCE AND REQUIREMENTS AFTER THE BROWNS FERRY FIRE.
DEFENSE-IN-DEPTH FIRE PROTECTION PRINCIPLES

1. PREVENTION OF FIRES

2. LIMITING FIRE DAMAGE THROUGH PROMPT FIRE DETECTION AND SUPPRESSION

3. DESIGNING PLANT SAFETY SYSTEMS SO FIRE WILL NOT PREVENT SAFETY FUNCTION FROM BEING PERFORMED.

STRENGTHENING ONE OF THESE AREAS CAN COMPENSATE IN SOME MEASURE FOR WEAKNESSES, KNOWN AND UNKNOWN, IN THE OTHER AREAS.
PREVENTION OF FIRES

ELEMENTS ....

CONTROL OF COMBUSTIBLES

CONTROL OF IGNITION SOURCES

PERIODIC FIRE INSPECTIONS

USE OF FIRE WATCHES
FIRE DETECTION AND SUPPRESSION

ELEMENTS ....

DETECTION

- ADEQUATE DISTRIBUTION OF FIRE DETECTION DEVICES

- PLANT PERSONNEL (ADEQUATE TRAINING ON HOW TO REPORT A FIRE AND WHAT ACTIONS TO MITIGATE ITS CONSEQUENCE)

- USE OF CONTINUOUS FIRE WATCHES WHEN AUTO FIRE DETECTION IS INOPERABLE OR NOT PROVIDED

SUPPRESSION

- USE OF AUTOMATIC FIRE SUPPRESSION SYSTEMS (HIGH FIRE HAZARD AREAS AND AREAS IMPORTANT TO PLANT SAFETY)

- MANUAL FIRE FIGHTING CAPABILITY (HOSE STATIONS AND PORTABLE FIRE EXTINGUISHERS)

- TRAINED ON-SITE FIRE BRIGADE
DESIGNING PLANT SAFETY SYSTEMS

ELEMENT ....

FIRE SEPARATION BETWEEN REDUNDANT SAFE SHUTDOWN FUNCTIONS (OPTIONS)

- EACH REDUNDANT FUNCTION IN SEPARATE 3-HOUR FIRE AREA, OR

- REDUNDANT FUNCTIONS WITHIN SAME FIRE AREA, ONE IS ENCLOSED IN A 3-HOUR FIRE BARRIER, OR

- REDUNDANT FUNCTIONS WITHIN SAME FIRE AREA, ONE IS ENCLOSED IN A 1-HOUR FIRE BARRIER AND AREA IS PROTECTED BY AUTO DETECTION AND SUPPRESSION, OR

- REDUNDANT FUNCTIONS WITHIN THE SAME FIRE AREA HAVE OVER 20 FEET HORIZONTAL SEPARATION WITH NO COMBUSTIBLES INTERVENING AND AREA IS PROTECTED BY AUTOMATIC DETECTION AND SUPPRESSION.
DEFENSE-IN-DEPTH PHILOSOPHY
CHECKS AND BALANCES

HYPOTHETICAL CASES

CORRIDOR - ONE TRAIN OF SAFE SHUTDOWN CAPABILITY PROTECTED BY A 1-HOUR FIRE BARRIER, AREA PROTECTED BY AUTO DETECTION AND SUPPRESSION

(CASE 1 - NORMAL EXPECTED FIRE MITIGATION)

- FIRE DETECTED BY DETECTION SYSTEM.
- FIRE BARRIER PROVIDES FIRE RESISTIVE (PASSIVE) PROTECTION UNTIL AUTO SUPPRESSION ACTUATES AND CONTROLS THE FIRE.
- FIRE BRIGADE Responds AND EXTINGUISHES THE FIRE.

(CASE 2 - FIRE MITIGATION WITH FIRE DETECTION INOPERABLE, CONTINUOUS FIRE WATCH POSTED IN THE AREA)

- FIRE DETECTED BY THE FIRE WATCH AND REPORTED TO CONTROL ROOM.
- FIRE BARRIER PROVIDES FIRE RESISTIVE (PASSIVE) PROTECTION UNTIL AUTO SUPPRESSION ACTUATES AND CONTROLS THE FIRE.
- FIRE BRIGADE Responds AND EXTINGUISHES THE FIRE.
HYPOTHETICAL CASES - CONTINUED

(CASE 3 - FIRE MITIGATION WITH FIRE DETECTION AND AUTOMATIC SUPPRESSION INOPERABLE, CONTINUOUS FIRE WATCH IS POSTED)

- FIRE DETECTED BY THE FIRE WATCH AND REPORTED TO CONTROL ROOM.
- CONTROL ROOM PROMPTLY ALERTS ON-SITE FIRE BRIGADE
- FIRE BARRIER PROVIDES FIRE RESISTIVE (PASSIVE) PROTECTION UNTIL FIRE CONTROL AND EXTINGUISHMENT CAN BE ACCOMPLISHED BY THE FIRE BRIGADE
- OFF-SITE FIRE DEPARTMENT ASSISTANCE REQUESTED TO ASSIST WITH FINAL FIRE EXTINGUISHMENT.
INSIGHTS

- MAINTAINING A BALANCE IN DEFENSE-IN-DEPTH IS IMPORTANT FACTOR TO PLANT SAFETY AND FIRE RISK REDUCTION.

  - MOST FIRE PRA's ARE BASED ON THE ADEQUACY OF THE FIRE PROTECTION (DEFENSE-IN-DEPTH) PROGRAM (e.g., FIVE METHOD ASSUMES DEFENSE-IN-DEPTH ELEMENTS ARE IN PLACE)

- FIRE WATCHES ARE IMPORTANT TO ASSURE PROMPT DETECTION OF A FIRE (WHEN FIRE DETECTION IS INOPERABLE)

- PASSIVE FIRE BARRIERS ARE RISK IMPORTANT WHEN FIRE SUPPRESSION IS EITHER DELAYED OR INEFFECTIVE.

- ON-SITE FIRE BRIGADES ARE CONSIDERED ESSENTIAL ELEMENT IN FIRE CONTROL AND EXTINGUISHMENT.
RESOLVING THE WEAKNESSES
IN CABLE TRAY AND RACEWAY FIRE BARRIER SYSTEMS

TECHNICAL ISSUES

- Capability of these fire barriers to perform their fire resistive function (may not have been qualified by fire test)
- Ampacity derating
- Inadequate installation practices

NRC ACTION PLAN

- PART I  Resolution of technical problems
  - Coordinate resolution of technical issues with industry
  - Clarify and develop a NRC position on fire endurance test acceptance criteria
- PART II  Conduct NRC sponsored small-scale fire test program (test of Thermo-Lag panels to ASTM E-119)
- PART III  Prepare inspection guidance for NRC field offices
FIRE BARRIERS USED TO SEPARATE SAFE SHUTDOWN FUNCTIONS WITHIN THE SAME FIRE AREA

SAFETY OBJECTIVE

ENSURE THAT A SAFE SHUTDOWN TRAIN IS FREE OF FIRE DAMAGE BY PROVIDING EITHER A 1-HOUR OR 3-HOUR FIRE BARRIER

FIRE BARRIER DESIGN OBJECTIVES

WHEN SUBJECTED TO A STANDARD TEST FIRE (ASTM E-119 OR NFPA 251), THE FIRE BARRIER SHOULD LIMIT THE HEAT TRANSFER FROM THE FIRE BARRIER MATERIAL’S EXPOSED SURFACE TO ITS UNEXPOSED SURFACE.

MAINTAIN ITS INTEGRITY DURING THE FIRE EXPOSURE AND HOSE STREAM TEST
RACEWAY FIRE BARRIER ACCEPTANCE CRITERIA

PROPOSED STAFF POSITION

- NFPA 251 (ACCEPTANCE CRITERIA USED AS BASIS)
- CRITERIA

LIMIT UNEXPOSED FIRE BARRIER SURFACE TEMPERATURE TO LESS THAN 250 °F RISE

FIRE BARRIER SYSTEM REMAINED INTACT DURING THE FIRE EXPOSURE AND WATER HOSE STREAM TEST WITHOUT DEVELOPING ANY OPENINGS THROUGH WHICH THE RACEWAY OR COMPONENT (e.g. CABLE) IS VISIBLE
RACEWAY FIRE BARRIER ACCEPTANCE CRITERIA

PROPOSED STAFF POSITION

- PROVIDES GUIDANCE FOR DETERMINING CABLE FUNCTIONALITY
- METHODS WHEN CABLES ARE NOT INCLUDED IN RACEWAY TEST SPECIMEN
  - COMPARISON OF INTERNAL TEMPERATURE PROFILES TO EQ AND LOCA TEST RESULTS
  - PERFORMING AIR OVEN TESTS OF PLANT SPECIFIC CABLES AT RATED VOLTAGE WITH MEGGER AND HI-POT TESTS
- METHOD WHEN CABLES ARE INCLUDED IN RACEWAY TEST SPECIMEN
  - CABLE INSULATION TESTING (MEGGER AND HI-POT CABLE TESTING)

BEFORE AND AFTER THE FIRE EXPOSURE FOR POWER AND CONTROL CABLES

BEFORE, DURING AND AFTER THE FIRE EXPOSURE FOR INSTRUMENTATION CABLES
CONCLUSIONS

- DEFENSE-IN-DEPTH APPROACH ASSURES THAT THERE IS SUFFICIENT FIRE-SAFETY DIVERSITY IN DESIGN AND OPERATION OF THE PLANT.

- THIS DIVERSITY CAN COMPENSATE FOR PROGRAMMATIC WEAKNESSES, INOPERABLE FIRE PROTECTION FEATURES, AND LIMITED FIRE PROTECTION FAILURES DURING ACTUAL FIRE.

- CREATES A INTERACTION BETWEEN FIRE PREVENTION, FIRE DETECTION AND SUPPRESSION AND PASSIVE FIRE PROTECTION THAT EFFECTIVELY REDUCES FIRE RISK.
Adequacy of Fire Protection Systems in older Nuclear Power Plants

Safety Objectives, Analysis Methods, Findings

Session #2, Paper No. 9

Author: H. Bittner, Siemens Power Generation Group (KWU)

Cologne, Germany
December 6 to 9, 1993
1 Introduction

The oldest German nuclear power plants currently operating were planned, built and licensed in the mid-sixties. At that time, civil engineering legislation included hardly any rules and regulations for planning and licensing besides those stipulated for non-nuclear power plants.

In the recent past, fire protection for German nuclear power plants has gained more and more importance. After completion of licensing, construction and commissioning of the most recent plants (Isar 2, Emsland and Neckarwestheim 2), nuclear regulatory authorities and experts appointed under the Atomic Energy Act focussed more attention on those events and event sequences which are relatively likely and can in principle also be attributed to the spectrum of loadings not specific to nuclear power plants. This covers in particular fires. Fires constitute a real danger, especially because their effects such as smoke and heat have a direct impact, and in nuclear power plants fires have also occurred repeatedly, causing serious damage in certain cases.

In this context, it is indeed legitimate to ask just how the problem fire protection is being dealt with in nuclear power plants still in service.

In Germany, verification of the adequacy of fire protection systems in older nuclear power plants is based on the following:

- Development of more detailed codes and standards and also developments in the fields of verification, approval and licensing.

- Fire protection developments in German nuclear power plant construction

It would certainly be of interest to describe these two developments in detail, but this would go far beyond the scope of my presentation.

This paper hence presents the methodology developed to date and used for our analyses as illustrated in a review highlighting the salient features of:
- Development of codes, standards and other documentation
- Developments leading up to today's state of the art equipment.
2 Historical Background and Development

2.1 Rules, Regulations and Other Documentation

In addition to the civil engineering legislation, corporate standards were prepared. The objective here was to clearly stipulate the special fire protection requirements which apply within NPP's.

The major corporate standards are listed below:

KTA 2101: Fire protection in nuclear power plants
KTA 2101, Part 1: General principles (in force since 12/85)
KTA 2101, Part 2: Structural components (in preparation)
KTA 2101, Part 3: Mechanical and electrical components (in preparation)

KTA 2102: Escape routes in nuclear power plants
KTA 2103: Explosion protection in nuclear power plants

These activities were complemented by a whole range of supplementary and supporting projects based in part on licensing requirements pertinent to nuclear facilities and in part on civil engineering considerations.

In this connection, I would like to mention the following activities:

1. GRS report "Consequences of Fires in Plants" (German title: Einwirkungen durch anlageninterne Brände) prepared as part of the German Risk Study (1979)
2. BMI project SR 144 "Status Report on Fire Protection, Associated Measures and Regulations in Nuclear Power Plants" (German title: Bestandsaufnahme brandschutztechnischer Gegebenheiten, Maßnahmen und Bestimmungen in Kernkraftwerken), prepared by the working committee Fire Protection in Nuclear Power Plants, 9/1981
3. BMI project SR 144/1 "Optimization of Fire Protection Measures and Quality Checks in Nuclear Power Plants Equipped with Pressurized Water Reactors"
(German title: Optimierung von Brandschutzmaßnahmen und Qualitätskontrollen in Kernkraftwerken mit Druckwasserreaktoren)

4. GRS intermediate reports (prepared for BMU) on the fire protection topics relating to KTA nuclear safety standards
   - Systems for HVAC, smoke and heat removal
   - Initiation and activation of active fire protection measures
   - Requirements on fire protection systems resulting from external event control concepts

5. Fire hazard tests performed on the German HDR (superheated steam reactor) test facility from 1984 to 1992

6. Final report of the working committee Fire Protection Measures in Nuclear Power Plants
   - Catalog of criteria and catalog of actions for fire protection measures in nuclear power plants still in service

7. BMI project SR 144/2 "Optimization of Fire Protection System Safety in Nuclear Power Plants Equipped with Boiling Water Reactors" (German title: Sicherheitstechnische Optimierung von Brandschutzmaßnahmen in Kernkraftwerken mit Siedewasserreaktoren), 4/91

8. GRS report "Reliability of Fire Protection Equipment" (German title: Zuverlässigkeit von Brandschutzseinrichtungen), 7/91

As it would be futile to attempt to give a complete list of activities performed in the field of fire protection, I would prefer to give just a few more typical examples:

- Development and validation of computer program for prediction of fire-induced pressure and temperature profiles in complex, multi-compartment systems

- Development of methods to account for fires in probabilistic safety analyses

- Specialist meetings and resolutions of the RSK and IAEA committees which have recently been focussing on the adequacy of fire protection systems in nuclear power plants still in service.
2.2 **State of the Art of Fire Protection Systems in German Nuclear Power Plants**

The following played a major role in the development of fire protection systems used in German nuclear power plants:

- Typical fires and associated damage
- Development of more detailed codes and other documentation as described in Section 2.1

**Typical fires and their consequences for fire protection concepts in nuclear power plants**

<table>
<thead>
<tr>
<th>Year</th>
<th>Event</th>
<th>Consequences</th>
</tr>
</thead>
<tbody>
<tr>
<td>1968</td>
<td>PVC fire in the switchgear building, Pleinting power plant</td>
<td>More consistent fire breaks for electrical equipment compartments/levels</td>
</tr>
<tr>
<td>1971</td>
<td>Turbine oil fire, Mühleberg nuclear power plant Turbine oil fire, Niederaußen power plant</td>
<td>Special fire protection measures for turbine generator units such as:</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Fire breaks between oil tank and lube oil pipe duct</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Fire trips</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Dedicated fire-extinguishing system for turbine generator unit</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Use of flame-retardant control fluid</td>
</tr>
<tr>
<td>1975</td>
<td>Epoxy resin lining fire in reactor well, Neckarwestheim nuclear power plant, unit 1</td>
<td>- Improvements to escape routes</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Fire-extinguishing systems for various areas inside the containment</td>
</tr>
<tr>
<td>1975</td>
<td>PVC cable fire of the redundant cable trays at Browns Ferry, units 1 and 2</td>
<td>- Provision of extensive fire breaks to separate redundancies, in particular for the cable routing areas (subdivision into redundancy sections)</td>
</tr>
</tbody>
</table>

Concurrent with the developments in the field of codes and standards, further significant progress was made in the development of fire protection measures in addition to the improvements initiated as a result of actual damage due to fires.
Developments from about 1977 onward:

- Extensive fire protection compartmentalization of the buildings and quality verification of the structural items
- Isolation of major fire loads
- Extensive separation of redundancies by means of fire breaks

Developments in the eighties:

- Introduction of pulse alarm technology for fire alarm systems
- Flame-retardant non-corrosive cables for areas inside the containment
- Extensive portable and stationary fire-fighting systems
- Smoke venting systems in the switchgear building
- Integrated oil supply system for the reactor coolant pumps
- Inert gas atmosphere in containment (BWR)

One crucial factor for specification of the fire protection systems required for nuclear safety is the importance of the protected items to plant shutdown.

On the basis of the above, the following can be established:

- Systems and equipment required for shutdown to hot, subcritical or cold, subcritical condition must be protected.
- One or more redundancies might have to be protected. In Germany, loss of more than one redundancy is not acceptable (single failure)
- It might become expedient to protect a backfitted safety system against existing safety equipment as if it was a separate redundancy entity.
  For such purposes, it does not suffice to have the operators lay cables to the safety system in the event of a fire or temporarily connect off-site cables to the safety system feeder even though this seems to be common practice in a number of VVERs.
3 Safety Objectives and Fire Protection Measures

The fire protection systems required are specified on the basis of risk potential and safety objective.

These safety objectives are as follows:

1 Nuclear safety
2 Limitation of radioactivity releases
3 Personnel protection
4 Loss Prevention
5 Availability

Typical fire protection measures and systems are assigned to safety objectives in Table 1.
### Table 1: Assignment of fire protection measures to safety objectives

<table>
<thead>
<tr>
<th>Measures to fulfill safety objectives</th>
<th>Safety objectives</th>
<th>Nuclear safety - shut down - maintain in subcritical condition - RHR</th>
<th>Radioactivity release limitation (§ 28.3)</th>
<th>Personnel protection</th>
<th>Availability - Intrinsic safety operation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Civil engineering measures for fire protection</td>
<td>Implemented safety measures</td>
<td>Four-fold redundancy</td>
<td>By example: - Enclosed radioactivity potentials - Pressure gradient</td>
<td>- Intrinsically safe operation - German Accident Prevention Regulations</td>
<td>- Intrinsically safe operation</td>
</tr>
<tr>
<td>Fire breaks for F80/F30 areas</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td>1</td>
</tr>
<tr>
<td>Spatial separation</td>
<td>2 (between redundancies)</td>
<td>1 (from fire loads)</td>
<td>4 (from fire loads)</td>
<td>2 (from fire loads)</td>
<td></td>
</tr>
<tr>
<td>Access routes for rescue operations</td>
<td>3</td>
<td>3</td>
<td>1</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>Special measures</td>
<td>Fire-resistant enclosures</td>
<td>2 + 3</td>
<td>3</td>
<td>1</td>
<td>2 + 3</td>
</tr>
<tr>
<td>Coatings, e.g. for cables</td>
<td>2 + 3</td>
<td>3</td>
<td>3</td>
<td>2 + 3</td>
<td></td>
</tr>
<tr>
<td>Systems engineering measures for fire protection</td>
<td>Fire detection systems</td>
<td>4 + 3</td>
<td>3</td>
<td>3</td>
<td>2 + 3</td>
</tr>
<tr>
<td>Fire extinguishers + hydrants</td>
<td>4 + 3</td>
<td>3</td>
<td>3</td>
<td>2 + 3</td>
<td></td>
</tr>
<tr>
<td>Fire-extinguishing systems</td>
<td>2 + 3</td>
<td>2 + 3</td>
<td>3</td>
<td>2 + 3</td>
<td></td>
</tr>
<tr>
<td>Smoke venting systems, smokeless zones</td>
<td>4 + 3</td>
<td>4</td>
<td>3</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>Minimization of inflammable items</td>
<td>1</td>
<td>1 + 2</td>
<td>1 + 2</td>
<td>1 + 2</td>
<td></td>
</tr>
<tr>
<td>Venting measures</td>
<td>3</td>
<td>3</td>
<td>3</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>Operational measures</td>
<td>Plant fire brigades</td>
<td>4</td>
<td>4</td>
<td>1 + 3</td>
<td>1 + 3</td>
</tr>
<tr>
<td>Training</td>
<td>4</td>
<td>4</td>
<td>1 + 3</td>
<td>1 + 3</td>
<td></td>
</tr>
<tr>
<td>Periodic testing</td>
<td>4</td>
<td>4</td>
<td>4</td>
<td>4</td>
<td></td>
</tr>
</tbody>
</table>

1 = Primary measures  
3 = Supplementary measures to "1" and "2"  
2 = Alternative measures for "1"  
4 = Other measures
4 - **Analysis Methods**

On the basis of the safety analyses performed by Siemens KWU on the service condition "fire", the fundamental approach listed below has proven to be expedient:

1. **Definition of the design principles for the service condition "fire", like**
   - hot/cold subcritical shutdown
   - no simultaneousity with other events
   - necessary grade of redundancy

2. **Specification of the safety systems as well as additional/alternative systems and assignment to safety functions(safety- and backup systems)**

3. **Assignment of safety-related components to dedicated compartments (Att.1)**

4. **Listing of the fire protection measures for each compartment on the basis of planning documents and of in-plant inspections**

5. **Quantification of the fire hazards by means of**
   - conservative subjective evaluation
   - manual calculations
   - computer calculations

6. **Assessment of potential equipment outages on the basis of safety objectives**

7. **Definition of necessary upgrading measures for**
   - nuclear safety considerations
   - other considerations

Remark: The assignment of typical fire protection measures to the safety objectives is shown in Table 1.
5 New Methods for Quantification

Probabilistic Analysis

Until now engineering activities in qualifying existing fire protection situations have been carried out on a deterministic basis. As far as I can see the situation probabilistic fire safety analyses will become more important in the future. These methods can also form a basis for decisions whether additional fire protection systems will worth it from the safety point of view.

Computer Calculations

Although the necessity for methods to calculate fire induced temperatures in buildings and to define necessary fire resistance rates of structural elements had been discussed very controversially in the past, especially in KTA working groups, these procedures seem to have a remarkable benefit nevertheless. Especially in order to clarify situations where there is a lack of experience to estimate possible effects of fires to safety relevant equipment or to sensitive structural elements.

Time has changed within the last 10 years. Foreseeing that the necessity for the elaboration of analytical methods will arise and that the experience in handling such tools will become indispensable, we have developed an own computer code, called TEMPW. TEMPW can simulate fires within an entire system of building compartments or only within individual compartments. It calculates the time dependent distribution of the amount of heat released by a fire within the entire system and the existing heat sinks. The associated mass and energy exchange within the system is determined on the basis of the corresponding balances and equations.

As most important processes are modelled in these complex equations: Different burning behaviour of fire loads (e.g. oil, PVC, other solid materials) and thus the release of heat in the fire zone. Flow of air and fume between the zones which exhibit different thermal conditions. Heat transfer processes by way of convection and radiation between the seat of the fire, compartment air and heat sinks. Thermal conduction processes in heat sinks, e.g. local temperature distributions in civil structures or sensitive plant equipment.
The TEMPW simulation model has already been used successfully to calculate the effects of various fires. Some of them are described below:

1st example:
A leakage-oil fire beneath the steam generators and reactor coolant pumps of a PWR had to be examined.
The fire results in a temperature and pressure increase in the zones comprising the steam generator compartment. After a short time the pressure relief devices between this zone and and another adjacent zone open and thus enable an exchange of air and fume within the entire containment.
As shown in Att. 2 for selected areas the average compartment air temperatures in the containment had been calculated.

2nd example:
An oil fire inside the pressure chamber of a BWR had to be calculated.
The analysis was strongly influenced by the fact that the resulting overpressure in the pressure chamber will step by step be reduced by connecting tubes with the condensation chamber. The heated fume-air-mixture displaces water columns in the condensation tubes and will be cooled in the water seal. The decreasing temperature of the air inside the pressure chamber after the fire's end is connected with a corresponding pressure drop. In this connection an overpressure in the condensation chamber can arise, which will be relieved by opening some flaps to the pressure chamber. This explained course of events is taken into account and calculated by the fire simulation model in detail (see Att.3).
6. Findings, Conclusion

Based on fire protection safety analysis carried out in different older German NPP’s it became obvious that usually the existing fire protection system has a relatively high state of the art.

The definition of the plant-specific improvements was characterized by the following aspects:
- use of technics which allow deficiencies (passiv before active measures)
- use of passive measures for nuclear safety aspects
- by improving measures for specific safety objectives at the same time the level of the other safety objectives also became upgraded

Examples for typically starting-points for upgradings:
-usually, depending on it’s location, fire load concentrations (cable- and oil areas) need case by case additional or better separation
-the principle of separating redundant safety trains is not consequently implemented
-for saving money it could be useful to realise a global protection of the fire load (e.g. cable coatings) and to accept at the same time existing small distances between redundant systems/components
-existing fire detection and fire fighting measures need to be completed and optimized

The overall objectives of fire protection, the concept-relevant aspects as well as the existing variety of fire protection measures and its selection criteria makes it clear, that especially for older NPP’s - plant-specific fire protection concepts need to be developed.

This may be a contribution and stimulation to positive future developments of comprehensive fire protection systems in older NPP’s.

END
### Att 1: Assignment of safety-related components to dedicated compartments

<table>
<thead>
<tr>
<th>Raum</th>
<th>AKZ</th>
<th>Komponente</th>
<th>Red./Schelbe</th>
<th>Bemerkung</th>
</tr>
</thead>
<tbody>
<tr>
<td>1116</td>
<td>TF415002</td>
<td>Arm. am Nachkühler-W</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>1116</td>
<td>TF415006</td>
<td>Arm. am Nachkühler-W</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>1116</td>
<td>TH409002</td>
<td>Nachkühler</td>
<td>0</td>
<td></td>
</tr>
<tr>
<td>1116</td>
<td>TH409001</td>
<td>Nachkühlpumpe</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>1116</td>
<td>TH409002</td>
<td>Sperrwasserdruckerhöhungspumpe</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>1116</td>
<td>TH409001</td>
<td>Arm. Saugleitung Flutbeh.</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>1116</td>
<td>TH409002</td>
<td>Arm. Saugleitung Flutbehälter</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>1116</td>
<td>TH409007</td>
<td>Arm. am Kühler</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>1116</td>
<td>TH409008</td>
<td>Arm. am Kühler</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>1116</td>
<td>TH409009</td>
<td>Arm. am Kühler</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>1116</td>
<td>TV055013</td>
<td>Arm. Probe nach Kühler</td>
<td>2</td>
<td>für TH-System nur Zu-Funktion</td>
</tr>
<tr>
<td>1116</td>
<td>TV055014</td>
<td>Arm. Probe nach Kühler</td>
<td>3</td>
<td>für TH-System nur Zu-Funktion</td>
</tr>
<tr>
<td>1118</td>
<td>TH015001</td>
<td>Arm. Sumpfabschluß</td>
<td>3</td>
<td>Nur Zu-Funktion</td>
</tr>
<tr>
<td>1118</td>
<td>TH045001</td>
<td>Arm. Sumpfabschluß</td>
<td>1</td>
<td>Nur Zu-Funktion</td>
</tr>
<tr>
<td>1119</td>
<td>TF115002</td>
<td>Arm. am Nachkühler-W</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>1119</td>
<td>TF115006</td>
<td>Arm. am Nachkühler-W</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>1119</td>
<td>TH101002</td>
<td>Nachkühler</td>
<td>0</td>
<td></td>
</tr>
<tr>
<td>1119</td>
<td>TH101001</td>
<td>Nachkühlpumpe</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>1119</td>
<td>TH101002</td>
<td>Sperrwasserdruckerhöhungspumpe</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>1119</td>
<td>TH105001</td>
<td>Arm. Saugl. Flutbehälter</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>1119</td>
<td>TH105002</td>
<td>Arm. Saugl. Flutbehälter</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>1119</td>
<td>TH105007</td>
<td>Arm. am Kühler</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>1119</td>
<td>TH105008</td>
<td>Arm. am Kühler</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>1119</td>
<td>TH105009</td>
<td>Arm. am Kühler</td>
<td>3</td>
<td></td>
</tr>
<tr>
<td>1119</td>
<td>TV055007</td>
<td>Arm. Probe nach Kühler</td>
<td>2</td>
<td>für TH-System nur Zu-Funktion</td>
</tr>
<tr>
<td>1119</td>
<td>TV055008</td>
<td>Arm. Probe nach Kühler</td>
<td>3</td>
<td>für TH-System nur Zu-Funktion</td>
</tr>
<tr>
<td>1121</td>
<td>TF215002</td>
<td>Arm. am Nachkühler-O</td>
<td>2</td>
<td></td>
</tr>
<tr>
<td>1121</td>
<td>TF215006</td>
<td>Arm. am Nachkühler-O</td>
<td>2</td>
<td></td>
</tr>
<tr>
<td>1121</td>
<td>TH208002</td>
<td>Nachkühler</td>
<td>0</td>
<td></td>
</tr>
<tr>
<td>1121</td>
<td>TH208001</td>
<td>Nachkühlpumpe</td>
<td>2</td>
<td></td>
</tr>
<tr>
<td>1121</td>
<td>TH208002</td>
<td>Sperrwasserdruckerhöhungspumpe</td>
<td>0</td>
<td></td>
</tr>
<tr>
<td>1121</td>
<td>TH205001</td>
<td>Arm. Saugl. Flutbeh.</td>
<td>2</td>
<td></td>
</tr>
<tr>
<td>1121</td>
<td>TH205002</td>
<td>Arm. Saugl. Flutbeh.</td>
<td>2</td>
<td></td>
</tr>
<tr>
<td>1121</td>
<td>TH205007</td>
<td>Arm. am Kühler</td>
<td>2</td>
<td></td>
</tr>
<tr>
<td>1121</td>
<td>TH205008</td>
<td>Arm. am Kühler</td>
<td>2</td>
<td></td>
</tr>
<tr>
<td>1121</td>
<td>TH205009</td>
<td>Arm. am Kühler</td>
<td>2</td>
<td></td>
</tr>
<tr>
<td>1121</td>
<td>TH205010</td>
<td>Arm. zur Kühlmitteleinigung/TA</td>
<td>2</td>
<td>Nur Zu-Funktion</td>
</tr>
</tbody>
</table>

---

**Note:**
- TF: Einspeisung --> Überflutung
- TH: eigene kleine Widerstände, Red.1
- Nur Zu-Funktion

![Image](image-url)
Att 2: Global calculation of oil fire inside PWR-containment

Diagram:
- Containment
- Pressure relief devices
- Reactor coolant pumps
- Steam generators
- Pressurized water reactor
- Leakage - oil fire

Graph:
- Temperature [°C]
- Time [min]
- Duration of fire
- Cooling phase
Att 3: Oil fire inside PWR-containment below steam generator

- a Containment
- b Pressure chamber
- c Pressure relief devices
- d Condensation tubes
- e Boiling water reactor
- f Condensation chamber
- g Oil fire

Graph showing pressure (bars) versus time (min).
Applied fire protection engineering;
A way to optimize the fire protection level

Sören Isaksson
Swedish National Testing and Research Institute

Prepared for the

Committee on the safety of nuclear installations
Principal working group No. 1 specialist meeting on

FIRES AND FIRE PROTECTION SYSTEMS
IN NUCLEAR POWER PLANTS

Cologne, December 6 - 9, 1993
Introduction

The discussion in this paper is mainly concerned with fire protection in older nuclear power plants.

In early design of nuclear power plants external events as fire was not identified as any considerable risk for nuclear safety. Regular industrial standards for fire protection installations were adopted and somewhat altered to ensure that radiological release was kept within the plant.

In the course of time the opinions about external events have changed and subsequently there is a great need for methods to determine the adequacy of the fire protection program in older plants.

Most older plants are built influenced by guidelines from ANS [2], USNRC [3] or the International Nuclear Risk Insurance Pool [4]. Since then, guidelines have been changed, and especially the regulations they refer to are outdated. The plants though remain mainly as they were designed.

In some countries this issue has been considered so important by authorities that back-fitting regulations have been issued requiring additional protection. One example is the USNRC CFR Part 50, appendix R, which followed the Browns Ferry accident [5].

The knowledge in fire science has increased considerably over the last two decades [1]. Better methods for measuring and analysis have made it possible to look into the basic physics and chemistry of combustion. Modelling of fires as well as whether forecasts have been improved by fluid dynamic science, especially turbulence modelling, and by computer development. This has caused today's trend in regulations to become performance based instead of prescriptive.

Three major steps must be considered to obtain optimal fire protection.

Proper regulations and testing standards

What is a proper standard? The answer is that it is not only covering the right type of system. It is also important to make sure that the testing procedure is representative for the application where the system is to be used.

Standards used for fire protection systems in nuclear power plants are almost only industrial standards. Sometimes there are smaller deviations from normally used standards, like requiring that a fire door should be opened and closed a number of times before being tested. On the other hand, to my knowledge there is only one country, Ireland, that has demands for smoke tightness in their normal testing standard. Smoke from door cracks affects escape for humans, makes manual extinguishing more difficult, and can sometimes also spread fire.

One important question is, how does ageing affect the system. Tests have shown that flammability properties of cables can change with time [10]. Also the vulnerability of components can be affected by ageing [11].

Premixed foam-water solutions have shown to be very short-living. Some brands loose half their effectivity in one year, and for some the time can be even shorter.
Fire protective coatings like intumescent or sublimation coatings are developed to perform well in the furnace test when subjected to a standard fire curve, like the ISO 834. If the coating is affected by a less, or a more severe environment it may not perform as well. If it is directly impinged by a flame the coating may be eroded by the flame. Ageing and vulnerability to chemicals is not tested either.

These were just a few examples of problems that should not be news to people deciding on choice of fire protection systems in nuclear plants. None of the issues are covered by standards, and there are numerous others that could be mentioned.

All regulations and standards have their limits. When using them, understanding of their background and context is sometimes of great importance. They can not be read "as the devil reads the bible".

Fire and fire protection is like nuclear safety a wide and complex field, touching nearly all parts of natural science. When retrofitting in an older plant it is usually not possible to use just one type of system, or to fully comply with a certain regulation.

No special requirements are made for example for systems replacing passive protection requirements, so called technical exchange. These regulations should be seen as minimum requirements to reach an acceptable level of fire protection in industrial applications. Reliability data show that sprinkler systems of standard design have a release rate of about 0.9 to 0.99 per demand [7], another source states that the probability of extinguishment ranges between 0.1 and 0.9 depending on actuation and system standard [8]. Fire alarm systems are also difficult to analyse since they can be of different types and sizes, the detection probability ranges between 0.2 and 0.9 depending of type and coverage [8]. These figures show how great the difference can be between a properly designed system and one that is not.

Apart from designing the fire protection system for radiological safety reasons, aspects like personnel safety, economic loss, production loss, political considerations and loss of public acceptance also must be taken into concern.

A fire in the spare part storage area of a plant could force the plant into outage for a considerable time. Many components of older plants are not in production anymore, and qualification of new parts takes a long time, others that are in production have long times for delivery.

How to choose the right protection method

First all the different demands on the system because of fire should be identified.

Shall it:
- detect the fire, and then what does the fire look like?
- extinguish or control the fire?
- be active or passive?
- be fast?
- prevent the formation or spread of smoke?
etc. etc.
Demands set by other conditions are also important and should be specified.

Shall it:
- be safe for humans?
- be safe for some certain type of equipment?
- be easy to maintain?
- be easy to install?
- have known ageing properties?
etc. etc.

All these demands limits the number of different ways of protection possible. The final solution is then usually decided from economic reasons among the acceptable systems.

**Inspection**

How do authorities in different countries assure that the fire protection level is sufficient? The answer is that the approach varies much from country to country.

Republic of South Africa for example relies to a high extent on the competence of the supplier and the owner of the plant. United States and Finland on the other hand demand that specific regulations should be followed for all fire protection features.

In Finland fire protection systems are inspected by the government authority every year and the building fire protection twice a year. In the U.S. on the other hand, both types of inspections are made every 36 months. For German plants the initial tests are performed by experts or by responsible authority depending on system type. For periodic testing either plant experts or external experts perform tests for example on fire alarm and extinguishing systems every year, and for other active fire protection systems every two years.

Apart from the frequency the quality of inspections is important. A person who is supposed to inspect and thoroughly evaluate the state of the fire protection program of a plant shall preferably be knowledgeable in all the following fields.

<table>
<thead>
<tr>
<th>Fire dynamics</th>
<th>Fluid dynamics</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ignition and flame spread</td>
<td></td>
</tr>
<tr>
<td>Passive systems</td>
<td>doors</td>
</tr>
<tr>
<td>- walls</td>
<td>fire protection coatings</td>
</tr>
<tr>
<td>- penetrations</td>
<td></td>
</tr>
<tr>
<td>Active systems</td>
<td>automatic extinguishing systems</td>
</tr>
<tr>
<td>- fire detection systems</td>
<td></td>
</tr>
<tr>
<td>- smoke control systems</td>
<td></td>
</tr>
<tr>
<td>Fire brigade training</td>
<td>Fire brigade equipment</td>
</tr>
<tr>
<td>Communication</td>
<td>Human behaviour under stress</td>
</tr>
<tr>
<td>Escape routes</td>
<td>Administrative controls</td>
</tr>
</tbody>
</table>

Obviously it is impossible for one person to master all these fields. But sometimes it is necessary to compare different blends of protection systems and evaluate if they are equivalent or sufficient. A total evaluation of a plant will also only be performed once, when it is commissioned. For following inspections it is usually
sufficient with only partial competence. For many older plants though this first evaluation can be questioned with today's knowledge.

Fire risk analysis

Risk analysis technique has shown that the original assumptions for nuclear power plant design with the design basis accident (DBA) as the main event contributing to core damage risk and redundant safety systems greatly reducing the chance of accident, are not valid [6]. Probabilistic risk assessments (PRA) performed on different plants show varying weights between internal and external events contributing to core damage, see figure 1. In figure 2 the variation in contribution from different external events is shown [9]. Geological siting as well as the large uncertainties in both data and modelling explains the varying fractions of core melt frequencies.

![Core damage graph](image1)

**Figure 1** Internal versus external contributors to core damage [9]

![Total core damage graph](image2)

**Figure 2** Contribution of external events [9]
One of the basic assumptions that has to be made when doing a traditional fire risk analysis is the fire frequency. Frequencies used are based on earlier events but it is very difficult to consider transient fire loads and ageing of installations. Therefore it is suggested to assume the fire frequency to be equal in all rooms, and then calculate the impact from damaging all the equipment inside the room. Then the time dependent probability for the fire to spread or adversely affect other rooms should be calculated, also taking in mind the effects of actuated extinguishing systems, and manual fire fighting.

Often as in [7] fire spread between fire areas is not considered since this phenomena has only occurred a few times in nuclear power plants. If looking at industry as a whole, failure of fire barriers is not so uncommon. Statistics are scarce though, since this type of information is not often registered. American statistics for low rise stores and offices show that for so called protected non-combustible constructions 6.6% of fires spread beyond floor of origin and 1.9% spread beyond room of origin but on the same floor [12].

A proposed working method

For many older plants the best way to obtain a clear picture of the fire protection level and to determine if it is acceptable, is not to try to comply with newer regulations or do extensive investments in protection systems. A more aimed method is described in short words below.

First define safety systems important for radiological integrity. A PRA/PSA type study should be performed on these systems. The study must cover objects, power supply and control system, and take common cause failures in consideration.

The PRA/PSA-study should be organised to enable the following.

i) deterministic evaluation of total fire damage within each fire area and the impact on radiological safety.

ii) deterministic evaluation of total fire damage in groups of adjacent fire areas and their impact on radiological safety.

From i) a ranked list of fire areas with potential for damage to safety systems is obtained. This list can be compared to an acceptance criteria to limit the number of rooms for the further study.

From ii) a ranked list of groups of adjacent fire areas with potential for damage to safety systems is obtained. This list can be compared to an acceptance criteria to limit the number of rooms for the further study.

A detailed study should be performed, for all fire areas, or groups of fire areas that could not be excluded during step B, to determine whether the potential fire risk is low enough or the methods for fire protection sufficient.
References


OECD Nuclear Energy Agency
Committee on the Safety of Nuclear Installations
Principal Working Group No.1, Specialist Meeting
on
Fires and Fire Protection Systems
in Nuclear Power Plants

UK APPROACH TO FIRE HAZARD ANALYSIS

J S COWLEY
Health & Safety Executive
Nuclear Safety Division
St Peter's House
Balliol Road
Bootle L20 3LE
England

October 1993
ABSTRACT

The detailed assessment of fire detection and prevention measures on nuclear power plants in the UK forms a major part of the internal hazards assessments which are carried out during the design and construction stages of a new plant, and also at subsequent periodic safety reviews during the operating lifetime of the plant.

The Regulatory Authority in the UK, the Nuclear Installations Inspectorate, published its revised Safety Assessment Principles for Nuclear Power Plants in 1992. These principles include a general requirement that fire detection and fire fighting systems of adequate capacity and capability should be provided where appropriate. In order to satisfy this, a fire hazard analysis should be carried out to determine the potential effects on safety systems and safety related plant; the need for segregation of plant; and the effectiveness of the fire detection and fire fighting systems to be provided. This paper describes the methods adopted in assessing the adequacy of the protection against fire on plant currently under construction, and also future designs.

Another important development in recent years has been the need to evaluate the fire safety provisions on older nuclear power plants. As part of these periodic long term safety reviews of the UK magnox reactors, the licensees were required to consider whether any additional improvements to fire zoning and equipment were needed, and also to re-examine the adequacy of the installed fire detection and suppression systems. This resulted in a considerable programme of improvements, and this paper outlines the regulatory approach to the assessment of the improvement programme.

Finally, maintenance of emergency preparedness to deal with any emergency, including fire, which may occur on a nuclear site is covered as a condition attached to the nuclear site licence. The arrangements for meeting this licence condition are briefly discussed.

1. Legislative Background

The main legislation which governs the safety of nuclear installations in the United Kingdom is the Health and Safety at Work etc. Act (HSW Act) and the associated relevant statutory provisions of the Nuclear Installations Act 1965 (as amended). Under these acts, no site may be used for the purpose of installing or operating any prescribed nuclear installation unless a nuclear site licence has been granted to a body corporate by the Health and Safety Executive (HSE). HM Nuclear Installations Inspectorate (NII) is that part of the HSE which has the responsibility for administering this licensing function.

There is also legislation specific to the fire hazard at prescribed special premises which, as well as nuclear installations, includes hazardous installations such as petrochemical installations. This legislation, The Fire Certificates (Special Premises) Regulations 1976, is confined to those measures necessary to mitigate the consequences of fire for the safety of employees on the site, but does not cover means for preventing fire nor matters relating to nuclear safety. These regulations therefore address mainly the adequacy of protected escape routes, the means of giving warning in case of fire, and the means of fighting fire, and will not be covered in this paper.
It is an important feature of UK safety legislation that, under the Nuclear Installations Act, absolute liability is placed upon the licensee for any injury to persons or damage to property. In addition, under the HSW Act, the licensee is responsible for the safe design and operation of nuclear installations and for ensuring, so far as is reasonably practicable, the health and safety of its employees and other persons. The duty of the NII is to ensure that the appropriate standards are identified, developed, achieved, and maintained by the licensee, and to regulate and monitor the safety of the plant by means of the powers granted under the site licence.

This legislative process is not intended to follow a prescriptive approach whereby the NII issues standards or codes of practice which must be observed. Instead, the licensee is required to develop and comply with its own standards, and the regulatory body (NII) then satisfies itself as to the adequacy and achievement of these standards. However, guidance is published by the HSE on the concept of "as low as reasonably practicable" (ALARP) in its document on The Tolerability of Risk from Nuclear Power Plants (ref 1). The HSE has also recently revised and reissued its Safety Assessment Principles for Nuclear Plants (ref 2), which are used by the NII in assessing the acceptability of proposed new nuclear plants, and are also used for comparing older plants against modern standards.

2. Assessment Principles

The safety assessment principles developed by the NII provide the basis for a systematic framework which can be used for the technical judgements which the assessors have to make. They are considered to represent the current view of good engineering practice, and we would not expect modern plants to have any difficulty in satisfying the majority of them.

Fire is recognised as a common cause initiating event which, without adequate protective measures, could render inoperable safety systems which are provided to reduce the risk from a nuclear plant, even where multiple (and diverse) safety trains have been provided.

There are three safety assessment principles which deal specifically with fire, these are:-

- Principle 141: fire detection and fire fighting systems of adequate capacity and capability should be provided where appropriate. They should be designed and located so that any damage they may sustain, or their spurious operation, does not affect the safety of the plant;

- Principle 142: to satisfy principle P141 a fire hazard analysis should be made of the plant to:
  
  (a) identify safety systems and safety-related plant;
  
  (b) analyse the potential for fire initiation and growth and the possible consequences on safety systems and safety-related plant;
  
  (c) determine the need for segregation of plant and the location and required fire resistance of boundaries to limit the spread of fire;
  
  (d) determine the capacity and capability of the fire detection and fire-fighting systems to be provided.
Principle 143: non-combustible or fire retardant and heat resistant materials should be used wherever practicable throughout the plant.

The objective of these principles is to ensure that:

- the probability of a large fire occurring is reduced to as low as reasonably practicable by minimising the quantities of combustible materials and sources of ignition;

- the safety systems and other essential support features are divided into fire zones, such that no single fire will give rise to a situation that could lead to a significant release of radioactivity or prevent the reactor from achieving a safe shutdown state.

These principles are supplementary to other key engineering principles which include general requirements such as:

- the need to consider all identified potential hazards

- plant layout

- defence in depth.

The defence in depth concept is also augmented by another key principles (no.62) which indicates a priority for plant sensitivity to any faults such that:

(a) a failure or maloperation should produce no significant operational response, or should produce a change in the plant state towards a safer condition;

(b) following a failure or maloperation, the plant should be rendered safe by the action of passive features or engineered safeguards which are continuously available in the state required to control the fault;

(c) following a failure or maloperation the plant should be rendered safe by the action of active engineered safeguards which need to be brought into service in response to the fault.

The implementation of these deterministic principles provides assurance that the potential hazard from fire has been reduced to as low as reasonably practicable.

3. Application to Modern Plant

Having outlined the main principles used in an assessment, it is now appropriate to consider how these have been applied to a modern plant, in particular the Sizewell 'B' design. The following paragraphs describe some aspects of decisions taken during the design stage which illustrate how particular assessment issues were resolved.

3.1 Fire Hazard Analysis

The licensing process in the UK for a new reactor design requires the assessment, by the NII of a pre-construction safety report (PCSР) which forms the eventual basis for the decision to
license the plant. For Sizewell 'B', a public inquiry was also held, during which a number of important safety issues were raised, including that of the potential vulnerability of the plant to the fire hazard. As part of its assessment of the PCSR, the Inspectorate raised a number of issues relating to the fire hazard safety case, in particular:

- the identification of those systems required to achieve, monitor and maintain a safe state;
- the required segregation of safety systems and the methods of achieving this;
- the reduction of the frequency of occurrence of fires;
- the reduction of the severity of any fires which may occur.

These issues resulted in a comprehensive fire hazard analysis being developed by the licensee which detailed on a room-by-room basis: details of the safety systems; fire loading; fire detection arrangements; fire extinguishing method; fire barrier penetrations; potential sources of ignition, etc. This information then provided the input to calculational methods for verifying the adequacy of principal fire barriers (a principal fire barrier is a barrier segregating redundant safety trains), by two methods:

- calculating the fuel load per unit floor area (the "Berry" method);
- calculating the temperature and duration of each room fire (the "Quintiere" method).

It was also considered necessary to support the second method by sensitivity calculations in a selected example room to examine the effects of various scenarios on the fire duration and maximum temperature reached. The scenarios included the effects of one, two or three fire doors remaining open and the ventilation system remaining operational. These conservative calculations gave further confidence to the adequacy of the 3 hour rating which was selected for the principal fire barriers.

Another important consideration was the potential for reactor coolant pump (RCP) oil fire to lead to failure of the RCP and steam generator supports. The concern was that the RCP oil lubrication system could fail, releasing oil onto the basement slab which in the event of fire, could degrade the integrity of the supports. Oil collection systems were installed to protect against this eventuality, but in addition a deterministic failure modes and effects analysis was required in order to demonstrate that no single failure mode would lead to oil accumulating on the basement slab.

Whilst the fire hazard analysis was accepted as part of the safety case for licensing the construction of the plant, it was recognised that as the design was developed, changes to the original specification were likely to occur. These changes, which could have the potential to reduce the in-depth defence against fire, have been the subject of continuing assessment throughout the construction phase. Prior to start up of the reactor a programme of plant walk-downs, some in the company of NII inspectors, is being undertaken in order to confirm that the design intent has been met.
3.2 Redundancy & Segregation of Safety Equipment

At an early stage, a decision was taken by the licensee to adopt a four safety train design for the segregation of redundant safety groups required to achieve safe "hot" shutdown. This was based on a deterministic approach which considered one segregation group to be lost in a potential fire, a second to be out on maintenance, and the third to fail on demand. Consequently, this, a further decision was taken to provide physical barriers of at least three hour fire resistance rating between redundant safety trains.

These two factors clearly contribute to safety by limiting the consequences, and spread, of a fire. Modern plants are, however, highly complex and it follows that whilst segregation by physical barriers is a desirable aim, its achievement in practice can be difficult. The Inspectorate has therefore paid particular attention to the integrity of fire barriers. Experience has shown that, whilst fire spread through imperforate barriers is highly unlikely, the potential for a fire to spread through penetration closures such as doors, cable or pipe conduit seals, heating, ventilation and air conditioning (HVAC) ducts and dampers is significantly greater.

As a result of these concerns for fire barrier reliability, the number of penetrations between segregated groups has been minimised, mainly by the re-routing of cables and HVAC ducting. The licensee was also required to produce a technical justification for any penetrations between adjoining redundant safety groups.

3.3 Fire Limitation

The philosophy of 4-way segregation has provided a considerable contribution to the defence-in-depth measures against fire. However, another important factor in fire prevention is the selection of materials used in the construction of the plant. It is well known for example, that experience with certain types of electrical cable insulation and sheathing (eg. PVC based) has shown the vulnerability of plant to fires developing in areas containing high densities of such cables. Combustible liquids such as fuel and lubricating oils can form another considerable source of potential fire loading.

In order to minimise the combustible inventory, the licensee selected fire retardant materials wherever practicable. This decision resulted in the development of cable insulation material which is resistant to fire initiation and propagation which, if ignited, produces less smoke and noxious fumes than the PVC equivalent. Tests on such materials have indicated that smoke and toxic gas production is of the order of 10% of typical PVC based insulation, and corrosive gas emission is about 1% of PVC insulation. Cable tray fire tests, using vertically bundled cables also served to demonstrate the self-extinguishing capability of the insulating medium.

Other examples of material selection are the choice of non-flammable or high flash point fluids for hydraulic actuators and associated supply systems when used in areas containing safety equipment; dry type electrical transformers with non-flammable cooling fluid are also used in safety sensitive areas.

These two examples illustrate how the safety principles relating to the use of non-combustible or fire retardant materials can be satisfied.
4. Application to Existing Plants

4.1 Regulatory Approach

The first generation of nuclear reactors (the gas-cooled magnox reactors) were originally expected to have a useful safe working life of between 20 and 25 years. Operating experience has given the operators confidence that this life could be extended, and in order to demonstrate this, they accepted the need to carry out an overall review of the safety of each station. This periodic or "long term safety review (LTSR)" has three objectives:

- to confirm that the plant is adequately safe for continued operation;
- to identify and evaluate any factors which may limit the safe operation of the plant in the foreseeable future;
- to assess the plant's safety standards and practices and introduce any improvements which are reasonably practicable.

The NII assessment of the first LTSRs for the oldest stations showed a pattern of ongoing issues which, it was concluded, might be expected to arise on all of the subsequent reviews. These were identified as key, or generic safety issues which were considered to be sufficiently important to require the licensees to give them priority attention, thus allowing improvements in safety to be achieved without awaiting the outcome of each station specific LTSR. One of these generic issues related to the fire hazard, and the licensees were required to:

- "consider whether any improvements to fire zoning and equipment were available and confirm the extent to which the installed system complies with modern standards".

4.2 Assessment Issues

The main difference in approach to the assessment of these existing plants was the limitation in capability to provide the level of safety system segregation which is possible on new plant such as Sizewell 'B'. The three basic objectives of the defence in depth approach can still be achieved. These objectives are (ref 3):

- preventing fires from starting;
- detecting and extinguishing quickly those fires which do start;
- preventing the spread of those fires which have not been extinguished, thus minimising their effect on essential plant functions.

However, increased reliance on fire detection and suppression systems, together with manual fire fighting techniques is necessary because the plant is not in all cases segregated into fire zones contained within fire resistant barriers, which also serve to segregate redundant safety trains.

The approach to fire hazard analysis for the older plants is similar to that for new plants in that the assessment should: identify plant which is required for safe shutdown; identify the location and rating of fire barriers; assess fire loadings; determine the adequacy of fire extinguishing
capability. In addition, the detrimental effects of smoke and the spurious operation of fire protection equipment also have to be analysed.

4.3 Engineered Improvements

The licensees developed strategies for carrying out the reviews of each station and implementing the subsequent improvements to segregation of essential plant and fire detection and extinguishing systems. It is not possible to detail all the findings of such a comprehensive review in this paper. The reviews of each station resulted in the implementation of an extensive programme of plant improvements, which included:

- provision of additional fire barriers, for example between adjacent safety systems;

- improved bunding of oil tanks;

- extensions to fire detection and extinguishing systems;

- provision of diverse fire detection and extinguishing systems in particularly sensitive areas, such as cable tunnels, where most cabling is of the PVC type;

- installation of ventilation and drainage systems in fire zone areas;

- repair of fire barrier penetrations.

In a small number of cases, major modifications to electrical switchboards were required to reduce the vulnerability of the plant to single failures. These modifications further improved the redundancy, segregation, and defence against fire.

5. Emergency Arrangements

The final element of the defence-in-depth approach is the establishment of emergency arrangements for extinguishing any fire on the site. This is covered by one of the conditions attached to the nuclear site licence, which requires the licensee to make and implement adequate arrangements for dealing with any accident or emergency arising on the site, including the possibility of a fire hazard, and that these arrangements are rehearsed at specified intervals. The arrangements are approved by the HSE. The emergency exercise demonstration takes place annually at each site and is observed by members of the NII. These exercises normally involve the emergency services such as police, fire and ambulance.

It is important that there is proper coordination between the nuclear plant and the local fire brigades, so that the latter understand the hazards associated with the plant. The responsibilities of the station fire teams and the arrangements for liaison with the local fire brigade therefore have to be detailed in site specific emergency handbooks. Generally, the arrangements provide for the senior fire officer to take charge at the scene of operations. Very close liaison is required between him and the “access control engineer” on the Station, who advises on radiological conditions or other technical matters. In view of the importance of this liaison, regular training sessions involving the station staff and the local fire brigade are encouraged.
6. Summary

This paper has reviewed the legislative background the control of nuclear power plants in the UK, and has described the approach to the assessment of the fire hazard.

The application of fire hazard analysis techniques to both modern and old plants has been discussed, and the methods of achieving defence in depth have been discussed.

Emergency arrangements to deal with any fires form an important feature of controlling fires, and the associated licencing requirements have been outlined.

7. References


(3) Fire Protection in Nuclear Power Plants (IAEA Safety Series No. 50-SG-D2 (Rev.1)).
FIRE PROTECTION DESIGN RULES for French Nuclear Power Plants

Maurice KAERCHER
Head of Division
ELECTRICITE DE FRANCE
SEPTEN
12, 14 Avenue Dutriévoz
69628 VILLEURBANNE CEDEX
Fundamental objectives of fire protection

• To ensure the security of people
  – measures for the evacuation of people
  – protection of fire brigades

• To maintain the performance of safety functions
  – The same fire cannot cause the unavailability of all systems performing the same safety-related function

• To limit damage to equipment which could result in long term unavailability
  – Provisions are taken to limit the cost induced by the plant unit unavailability or the repair of an equipment
FIRE PROTECTION DESIGN BASIS

• Main principles
  – Fire protection shall be design to ensure adequate safety that means
    » Integrity of the reactor coolant system
    » Capacity of safe shut down of the reactor and of maintaining it in the safe shut down conditions even after an accident
    » Capacity of limiting off-site radioactive release to acceptable limits
  – Only one fire postulated at the same time anywhere in the plant

• Common mode failure prevention
  – Redundant safety system should not be affected by the fire
    » Passive means to separate redundant safety systems
    » Division into fire compartments and fire cells
FIRE PROTECTION DESIGN BASIS

• Detection
  – Fast detection
  – Reliability
  – Fire localisation
  – Trigger alarm and eventually automatic actuation

• Fire fighting
  – Fixed or mobile
  – Smoke control system
VULNERABILITY ANALYSIS

- To ensure that the requirement concerning the separation of redundant systems is met a vulnerability analysis is performed at the end of the studies

• This analysis is achieved in each fire compartment to detect connection between
  - Mechanical or electrical equipment belonging to two redundant trains of the same system performing a safety function
  - Mechanical or electrical equipment belonging
    » To one train of a system performing a safety function
    » And to the systems required to operate the same system of the redundant train
  - Electrical cables eventually affected by unselectivity of protections in case of short circuit

• Conclusion
  - If a potential common point remains, it will be necessary to install passive protections
Fire prevention

- Potential fire hazards limitation
  - Choice of materials as non flammable as possible
  - Lay out avoiding the proximity of hot walls (T > 100°C), conducts conveying flammable fluids and electric cables
  - Fire load evaluation
    - Evaluation of fire duration (see diagram)
    - Statistic curve of fire load (see diagram)
  - Occasional fire load management
Fire spreading limitation

- Splitting up into fire compartments and fire cells directly connected with
  - Location of safety-related equipment (cables and components)
  - Location of main escape routes intended for security (evacuation of personnel and fire fighting brigades)
FIRE PROPAGATION in fire cells

Fire cell 1

Fire cell 2

air input

air output
Construction provisions

• Fire resistance rating for separation of redundant trains: not less than 1 H. 30

• Fire rated doors: mechanical and thermal qualification tests
  – mechanical tests: 400000 cycles of opening and closing
  – fire test: mechanical resistance, thermal insulation, flame proof, no emission of flammable gases from the unexposed face (cotton test)

• Wall penetration: thermal and leak tightness qualification tests

• Cables protection qualification test
  – Initial conditions: preheated to simulate the heat produced by the electric current
  – Functional test criteria: the cables are in operation during the test
REponce DES DETECTEURS

DETECTOR TESTS

SCHEMA 1

Mousse
PVC (CABLES)
Huile (OIL)
Cables
Oil
Papers
Huile + Calo
Carton (Hard Papers)
Générateur

<table>
<thead>
<tr>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
<th>8</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>2</td>
<td>3</td>
<td>4</td>
<td>5</td>
<td>6</td>
<td>7</td>
<td>8</td>
</tr>
</tbody>
</table>

V

E
LAY OUT DESCRIPTION

- Main transformer from Montereau (140 MVA)
- Oil confinement
  - Bunker around the body of the transformer
  - Leaktight retention pit on the lower part
- 2 sets of spray ramps
- Water screen separation
TESTS DESCRIPTION

- Fire in the upper part
- Fire in the lower part
- Transformer filled with water and a 10 cm oil layer
- Variation of the spray flow
- Pure water or water with foam
TESTS RESULTS

• Fire in the upper part
  – pure water is not efficient
  – fire extinguishment with doped water: 10 to 20 l/min/sqm in less than 2 min
  – full extinguishment with foam

• Fire in the lower part
  – fire extinguishment with pure water: 20 l/min/sqm in less than 1 min
  – the bunker limits the power of the fire
Session #3

Operating Experience and Safety Issues

Mr. Liemersdorf, Chairman
WHAT TO DO IN CASE OF FIRE?

Cologne, 5 - 9 December 1993

E. GORZA
WHAT TO DO IN CASE OF FIRE?

1. Introduction

At any industrial site and, therefore, also at nuclear power plants, fires are a potential hazard. Accordingly, the fire risk must be identified correctly and make the subject of a risk analysis.

Once this analysis has been completed, the passive means of fire protection, e.g. fire areas, specially selected electric cables, lube oils and construction materials, and the active means of protection, e.g. fixed fire fighting installations, fire detection systems, fire extinguishers, hydrants, are determined.

These means are implemented first of all to prevent any start of a fire and its propagation and, secondly, whenever possible, to extinguish the fire or at least confine it or keep it under control.

The fundamental aims of fire protection in a nuclear plant are:

- on the one hand, safe shutdown of the reactor, and
- on the other hand, preventing uncontrolled production and release of radioactivity to the environment.

These aims must be achieved even if the fixed fire fighting installations do not come into action.

Despite all the passive and active means having been implemented, "fast and appropriate" response still remains essential, since the first seconds after a fire has started are the defining ones. Therefore, the intervention procedures must be at hand so as to know which automatic responses will be carried out, and which fire fighting resources are available close to where the fire has been detected.

With this in mind, in collaboration with the Belgian nuclear plant operators, a file containing these procedures and information has been created. This file has been called RGI (for Récapitulatif Général Incendie); roughly translated we could call it "Fire General Summary".

Such a file has recently been drawn up for one of the seven Belgian nuclear power plants, which are all of the pressurized water type:

- one group of four units is installed in Doel, North-West of Brussels; these units together produce 2720 MW;
- one group of three units is installed at Tihange, South-East of Brussels, these units together produce 2770 MW.

The units that were first built, i.e. two at Doel and one at Tihange, came on stream around 1975, and the other four units between 1983 and 1985.

During the years 1983-1985 the fire protection of the first three units was reviewed to upgrade it to the safety level of the more recent units.

From this presentation of the nuclear generating system in Belgium, we can understand how fire protection became different between the power plants, and even between different units of a same nuclear plant.
2. Why was such a fire file created?

A tool such as the RGI file can help taking the right decisions when faced with a start of fire.

Whilst the risk of fire in a nuclear power plant is considerable due to the amounts of combustible products present and their proximity to sources of ignition, the resources applied towards confining, detecting and extinguishing a fire that has just started are in general not less considerable.

Listed below are the fire protection resources available in a 1000 MW power plant put into service in 1983:

- 170 fire areas
- 114 fixed protection systems (deluge and sprinklers)
- 550 fire doors
- 440 fire dampers
- 2 detection control centers
- 350 detection loops
- 3650 detectors
- 16 outside hydrants
- 132 hydrants
- 245 hose reels
- 491 extinguishers

Such figures indicate two things: firstly, the safety measures against fire have far from been neglected and, secondly, on account of the sheer numbers of equipment, a tool is necessary to manage it all.

It is important for what follows that the meaning of the word "management" be absolutely clear.

In the case before us, management is understood as:
- the description, identification and location of all this fire equipment, and
- the periodic inspection and maintenance of this equipment.

This management makes it possible to rapidly acquire a good knowledge about the installations and about their interaction, and to have the certainty that equipment in proper operating order is present at the location of the fire.

The management tool for this has been developed by means of a CAD-type software.
3. **RGI file presentation**

3.1. **Introduction**

Without going back in the history of this file, the point has to be stressed that it underwent several development stages before it reached its present stage. Each of the stages brought improvements on the previous ones, and leaves the possibility of customizing the tool to the user’s requirements.

3.2. **General presentation**

The RGI file comprises:

A colour printout of the various RGI sheets, available in the unit control room, comprises a number of A3 format sheets, with at least as many sheets as there are fire areas in the generating unit.

To have access to the various sheets of the file, a single access key is needed: the fire area identification number. The reason for choosing this key lies with the fact that a mimic board dedicated to fire safety is provided in the control room, showing the alarms generated by the detection and protection systems. On this display, the various buildings and their fire areas are represented by separate blocks.

Each fire area is identified by a warning light having its own number.

This light groups the alarms regarding the fire equipment present in the fire area, such as fixed protection installations, detection, fire damper positions, etc.

Since this mimic board existed, it seemed most appropriate to use the fire area number as an access key to the file, though other keys could be provided if more suited to a particular application.

3.3. **Presentation of a sheet of the file**

First, it is important to find out why the RGI, and particularly each of its sheets, are instrumental towards not losing any precious time when a fire alarm appears in the control room.

After this, we will review to what extent this file, can make management of inspection and maintenance of the installations easier.

3.4. **Intervention support**

If, in the control room, an alarm appears on the mimic board, the operator who is present responds.

He opens the RGI file at the page number indicated by the warning light, and thus obtains a sheet that gives him all the information concerning the fire safety in that particular fire area.
3.4.1. General format

All the sheets are in A3 format, since this format is easy to handle, classify and carry to the location of the fire.

Should a A3 sheet not be large enough to contain all the information regarding a fire area, it may be extended up to a total of three sheets.

Generally, at the top of the sheet is given the usual information such as the index, date, etc.

In the top right corner is shown the access key to the file, i.e. the fire area identification number.

Also shown are the initials of the building in which the fire area is located, the rooms that belong to that fire area, and at which elevation of the building it is situated.

For the sake of clarity, it is necessary that we should break down this A3 sheet into six dedicated windows.

The purposes of these dedicated windows being self-evident, we will not discuss each of these windows in detail. Rather, let us point out just some particularities.

3.4.2. Window 1

The first window comprises the information regarding the location of the fire area within the site, essentially useful to the outside emergency services.

This window shows:

- the location of the building within the site;
- the location of the fire area within the building;
- the location of the outside fire fighting equipment, e.g. outside hydrants;
- the indication of North, useful regarding the direction of the wind.

3.4.3. Window 2

The second window gives the elevation of the fire area within the building, and the identification numbers of the adjacent fire areas.

3.4.4. Window 3

The third window specifies the automatic actuations that will take place when a fire has been detected.

The window indicates:

- the automatic actions on ventilation equipment, these being the stopping or starting of fans and the actuation of fire dampers.

- the same philosophy has been adopted as regards automatic actuation of fixed fire protection systems by the detection system.
3.4.5. Window 4

This fourth window supplies all the information related to the active systems installed for signalling the start of a fire.

It shows:

- the diagram of the fire detection networks within the fire area, with each detector identified according to the information available at the detection control centre. When a detection network extends beyond the fire area, the indications are given of the adjacent room and fire area entered by the network;

- the same philosophy has been adopted concerning the network of fire alert push-buttons;

- the telephone sets within the fire area and in its immediate surroundings, which are identified and located, and are also part of the means of raising the fire alert;

- a network of closed-circuit TV cameras, providing indications of fires which start in locations that are hard to access under normal operating conditions, e.g. in the cells of the primary pumps in the reactor building.

3.4.6. Window 5

The fifth window comprises all the data related to the active fire fighting equipment:

- the location of the hydrants, hose reels and portable extinguishers. Notice the fire fighting equipment available in the close vicinity of the fire area, shown by symbols. Equipment available at a greater distance from the fire is shown only by the elevation, the room and the type available. This way of showing the equipment avoids overcrowding, and thus obscuring, of the maps.

- the location of the fixed fire fighting systems such as deluge, sprinkler, halon or carbon dioxide. For clarity, only one symbol is used, the type of the system being discriminated by a single letter. The location where the cut-off station or valve of the system is located is indicated in the reference section, accessible with a key which, again, is the fire area number.

- the various manual actions possible on the fixed fire fighting systems.

3.4.7. Window 6

This window provides the information regarding the passive installations involved in confining a fire, as well as regarding the risk faced in the fire area.

It shows:

- the limits of the fire area, and the identification number of the adjacent fire areas;

- the identification of the risk present (electrical, radiation, suffocation, ...);

- the safety classification of the room, its belonging or not to the nuclear zone, its fire risk level according to the Belgian regulations;

- the signalling of the normal and emergency exits;

- the location of the fire dampers, smoke exhaust dampers and fire doors of which the automatic actions are represented in window 3;

- the identification of a number of industrial doors (anti-flood and industrial safety);

- the various manual actions possible on the ventilation systems.
3.5. **Inspection and maintenance resources**

The examples that follow, taken from a few routine situations in the daily operation of a plant, can demonstrate how the RGI, apart from being a tool for location, identification and description of fire equipment, also can be extremely useful in the management of the inspection and maintenance requirements of the same fire equipment.

3.5.1. **Example 1 : Periodic inspection of extinguishers**

The organisation and implementation of inspection of extinguishers, for instance, may be based on the RGI file.

Indeed, a roundsman, or perhaps an inspector from an outside approved inspection organisation in charge of inspection of the condition of the extinguishers can carry with him a copy of an RGI file sheet in order to check whether the correct types of extinguishers are physically present at the locations indicated on the sheet. Additional marks entered on the sheet could indicate whether the equipment is in proper operating condition or needs to be replaced.

Information technology may even go further in rationalizing the work: by integration of abstracts of a number of sheets into one document that represents, for instance, an entire floor, the roundsman or inspector can carry out his work for the entire floor without having to consult a variety of documents.

3.5.2. **Example 2 : Unit tests on fire dampers**

The same computerised support (as in above example 1) can be provided for the unit tests on fire dampers, when checking damper operation and damper position signalling to the control room of the unit.

3.5.3. **Example 3 : Detector maintenance**

The replacement of a detector can also be organised with the RGI file.

Indeed, if a detector is faulty, it is identified at the control room by its number and by its location (room). Since these items of information are recorded in the RGI file, the faulty detector can be found easily on the actual premises provided there exists a database which can tell the relation between the room and the fire area to which that room belongs.

There is no point in giving many further examples, since these could but confirm the possibilities of the RGI file as a support tool for inspection and maintenance.
4. **Evolution of the file, and possible extensions**

As already mentioned at the start of the presentation, the RGI underwent several development stages that each brought improvements on the previous stages, though we are aware there will always be scope for further improvement. Information technology keeps opening up new possibilities, and plant operators keep specifying new requirements. As a result, improvements to the file and creation of extensions to it are being envisaged.

Without entering into detail, these could come in the following directions:

- the computer connection of the RGI to a database, so that equipment could be managed with barcodes;

- the electrical connection between the mimic board and the RGI so as to supply directly to the operator in the control room a colour printout of the RGI sheet relating to the warning light which has lit up on the mimic board;

- the addition, for each fire area, of precise instructions or details regarding unavailability of equipment or access routes;

- the improvement of fire equipment management possibilities;

- etc.

5. **Conclusions**

As a conclusion, it can be said that man has always endeavoured to keep control over fire, but has never fully achieved it, and probably neither will information technology. Since information technology, on account of the many possibilities it offers, can definitely be a help, it should be taken on board so that we may improve our effectiveness in relation to a risk as serious as fire.
IMPROVEMENTS IN FIRE SAFETY AT LOVIISA NPP

Abstract

The fundamental objective of fire protection is to maintain safety functions during and following any fire. This means that the reactor can be shut down and maintained subcritical, the plant cooled down, the residual heat removed and any release of radioactive substances prevented.

Fire protection can be provided by plant lay-out considerations, physical separation of the equipment and fire barriers between important items. Fire detection arrangements, fire fighting systems and operational fire fighting provisions additionally increase fire safety.

The original passive fire protection concept at the Loviisa NPP (PWR, type: modified VVER440-V230) was inadequate due to the out-dated functional lay-out design of the plant (original and modified), deficiencies in physical separation of the equipment and lack of fire zones for systems and components.

Many improvements in fire safety have been accomplished but some are still under consideration. After the commissioning, the most important fire safety problems concerned residual heat removal through the secondary circuit and electrical AC-power supply. For ensuring the main safety functions, many structural improvements and also lay-out modifications concerning safety systems have been completed, as well as comprehensive fire extinguishing systems and fire fighting arrangements added. These will be described in the presentation.
A special fire-PSA has already given, and will hopefully in the near future give, answers to the remaining fire safety questions.

1 Ensuring residual heat removal through the secondary circuit

Original status

A nuclear reactor generates heat, so called residual heat, for a long time after shutdown. Its removal from the reactor is essential to prevent core damage. In a PWR type plant like Loviisa the reactor core is continuously cooled by water circulating in the primary circuit. From primary circuit water, heat is transferred to water in steam generators' secondary side from where heat is transferred away via secondary circuit steam lines. Water to the steam generators' secondary side is fed by means of the feedwater system. For transients, there is the emergency feedwater system. In addition, heat removal from the primary circuit by the secondary circuit presupposes a reliable functioning of the isolation and safety valves of the steam lines.

At the Loviisa nuclear power plant, the turbine hall is exceptionally important for plant safety due to the outdated functional lay-out design of the plant (see the original VVER440-concept). Essential parts of the systems required for the removal of reactor residual heat have been placed in the open space of the turbine hall which is common to the plant units. These parts include items such as

- feedwater pumps
- emergency feedwater pumps
- isolation and safety valves of the main steam lines
- valves of the feedwater lines.

The valves mentioned shall be functional irrespective of which ever of the feed water systems feeds the steam generators.
Also the plant control building, including the control room as well as important electrical and instrumentation rooms are in connection with the turbine hall. Certain electric connections important for the plant electricity supply go through cable tunnels which are housed in the turbine hall or in the open space of the turbine hall.

Improvements

The first attempts (in 1980) to improve fire safety in the turbine hall were

- the isolation of load bearing steel structures of the turbine hall and the control building
- the installation of sprinkler systems to protect the turbine hall and the oil systems of the turbosets
- adding the option to feed water to steam generators by the equipment of the primary circuit make-up water system.

A reassessment project of turbine hall fire safety started in 1985. The project aimed at ensuring residual heat removal particularly in connection with a potential major turbine hall fire in which the systems usually removing residual heat could be lost.

As the result of the project the following measures have been taken:

1. The so called B-line firewall of the turbine hall was constructed (Fig. 4). The firewall separates steam line safety and isolation valves and feed water line valves from the turbine hall into an individual fire zone to protect them against impact of turboset oil fires.

2. Both plant units were provided with new additional emergency feedwater systems (Fig. 4). A complete independence of a turbine hall fire was the design
basis of this system. The system was commissioned to operation at the beginning of 1990.

Furthermore, the new systems give improved protection against a station blackout.

This separate, independent system to ensure feedwater supply is placed in its own building. This back-up emergency feedwater system is schematically presented in Fig. 1. The system has an own water supply, two diesel-driven pump units and own separate pipelines direct to the steam generators of both units. If necessary, water can be fed both to Loviisa 1 and Loviisa 2 by using either diesel pump. At both plant units, water is fed to four steam generators via the drainage nozzles in the bottom of the steam generators. The required water is drawn from the already existing pure condensate storage tanks near which a separate back-up emergency feedwater pump station has been built.

The new system has been taken into account in the plants' PSAs. The improvements' diminishing effect on the probability of severe core damage is significant.

3. Measures relating to the ensuring of the environmental durability of rooms and components important to the removal of residual heat have been completed (new cable ways and tunnels and new smoke extraction hatches in the turbine hall).
2 Ensuring AC-power in the case of main transformer fires and other fires

Original status

After a large transformer fire in a conventional transformer station, it was necessary to study the probability and impact of a similar event at nuclear power plants.

At the Loviisa nuclear power plant, the diesel building common to both plant units is located in the immediate vicinity of the main transformers (Fig. 2). Both plant units are with four stand-by diesel generators to assure electric power supply to systems and components important to safety in case off-site power is not available. Both plant units have been connected to the national grid through two 400 kV main transformers. In addition, the units have one 110 kV grid connection through a plant transformer in common (Fig. 3).

When the consequences of a main transformer fire were estimated, the possibility had to be considered that both the 400 kV and the 110 kV grid connections are lost due to the fire and several diesel generators are simultaneously damaged as well. It was estimated that the probability of a transformer fire is so high that it was to be taken into account.

Improvements

- A 110 kV/6 kV plant transformer was transferred outside the postulated fire area in the 110 kV switchyard. At the same time, some modifications were carried out which make it possible to use one diesel generator to supply electricity to the other plant unit (Fig. 4). In addition, the supply is backed by two gas turbine units at the plant site.

- The supply of clean inlet air to the diesel generators was ensured by constructing new inlet air channels farther away from the postulated fire area.
Furthermore, the structural fire protection of the diesel building was improved.

- The front wall of the transformer bunkers was raised.

- Structural improvements of the diesel building and turbine hall were made.

- The fire insulation of the supply cables of the diesel generators was improved in the lower parts of the turbine hall.

- The hydrogen bottles of the generator cooling system were moved from the postulated fire area to a safer place (Fig. 4).

- An automatic sprinkler system for all transformers (12 pcs) in the main transformer yard was constructed.

- The plant was provided with a 20 kV grid connection from a hydropower plant.

3 Other fire studies and improvements

A special fire risk study has been carried out for fires inside the containment. Detailed numerical fire simulations have been done for the steam generator space and an electrical room. The results and conclusions will be dealt with together with the effects of other possible fires in the context of the fire PSA.

The impacts of small transformer (6/0.4 kV) fires have also been studied. Only minor improvements have been necessary.

The effects of an off-gas system coal filter fire were estimated considerable. A detailed study proved, that the probable radiological consequences are tolerable. However,
the big coal filter columns have been equipped with provisions for nitrogen gas supply.

For improving the reliability of fire water systems there are plans to construct a new, separate fire water pumping station to supply sprinklers and fire water net, consisting of
- pump station with 3 diesel driven fire water pumps
- new water tanks
- new supply lines for the main ring
- valve stations for fire brigades.

The PSA-study (by IVO) for Loviisa 1 NPP consist also floods, earthquakes and fires. The comprehensive fire-PSA work has begun in 1988. The study is not yet completed but a lot of results are available. On the basis of identified risks also many improvements are completed and several are still under the design e.g.
- additional sprinklers for cables and oil systems of pumps and control valves
- fire insulations and fire resistant coatings for certain cables.

4 Operational fire fighting and procedures

The operational fire fighting abilities of the fire brigade and other fire fighting troops have been remarkably improved. Presently there are e.g. a professional fire foreman and two professional firemen in the shift. Also the operational fire fighting equipment and other provisions have been improved.

The evolution of the procedures has also been important. The periodical inspection and testing programmes have been further developed. Also many detailed procedures for testing the fire fighting systems and other items have been revised and completed. Fire protection training is a regular part of the training program for the plant personnel. The large fire drills are arranged regularly with fire brigades of the neighbouring region. The fire protection procedures for main-
tenance and repair work and work permit practices are of great importance.

For managing the plant in case of fires a large number of emergency procedures has been compiled.

5 Conclusion

Several improvements to ensure the safety of the plant in case of fires have been carried out. The most important of these have been improvements in ensuring residual heat removal and AC-power. Remaining fire safety questions will be reviewed in the light of the final results of the fire-PSA. We hope that by the fire-PSA, it is possible to assess and to solve the remaining fire protection problems in a sound way.

Strict and well-qualified practices in maintaining fire protection measures are necessary for the duration of the plant's life. Vigilant awareness among the plant personnel of potential fire risks is vital for the safety of the plant.
Fig. 7. Diagram of the back-up emergency feedwater system of the Loviisa plant.
Figure 3  Loviisa 1 and 2, Supply of service power 110 kV-BT03-Lol, Lo2 (BT05)
Figure 4  General layout of the plant site
Main Findings from Reviews
of the Fire Protection Measures in Older German NPPs

Dr. M. Röwekamp / Dr. T. Riekert
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Cologne

OECD-Specialist Meeting on
Fires and Fire Protection Systems in Nuclear Power Plants

Cologne, Germany, 6th to 9th December, 1993
Contents

1 Introduction 1

2 Fire protection concept of German NPPs 2

3 Results of plant specific reviews 4

3.1 Separation of fire compartments 4

3.2 Fire detection and stationary fire extinguishing systems 5

4 Consideration of manual and automatic actuation 7

4.1 Advantages and disadvantages of automatic actuation 8

4.2 Incidents involving fires and fire fighting systems 9

4.2.1 International experience 9

4.2.2 German experience 10

4.3 Current actuation practice in German nuclear power plants 10

5 Conclusions 12

6 References 13
1 Introduction

German Nuclear Power Plants (NPPs) now under operation have been constructed in different plant generations.

In comparison to the development of the plant and safety technic with their continuity the development of fire protection measures shows considerable steps between the different plant generations. In the first part of the presentation, the main aspects of these steps of development for fire protection in NPPs will be shown. Fire safety in NPPs has been influenced on the one hand by the development of the conventional fire protection requirements for buildings and industry. On the other hand, the fire protection concepts for NPPs have taken more and more into account the safety related requirements which are a main aspect of this paper.

Since the constructional phase of the plants the fire protection measures of older German NPPs have been upgraded extensively. Nevertheless, those procedures were not very homogeneous at all. Therefore, special reviews of the fire protection measures have been accomplished in the last years. These reviews are basing on a deterministic approach. For some of the plants the investigations are not yet finished. The second part of the presentation gives a short survey about the results in general, including the following aspects:

- Separation of fire compartments with respect to safety systems redundancies,
- Fire detection systems,
- Fire extinguishing systems.

As one important result of these reviews, it was found that the time delay between a fire alarm and the actuation of fire fighting is too long for lots of fire situations. This fact caused an intensive discussion concerning the possibility of actuating automatically more of the stationary fire extinguishing systems. The problem of the automatic actuation of stationary fire extinguishing systems will be mainly discussed in the third part of the presentation.
2 Fire protection concept of German NPPs

In the past, fires have been among the worst incidents occurring in nuclear power plants. These fires do not only cause the rather conventional risks to life and property, but may also endanger the overall safety of the plant. Therefore, fire protection in NPPs has at its prime task to guarantee the following safety functions in case of fire:

- reactor shutdown,
- maintaining long-term subcriticality,
- residual heat removal (RHR),
- prevention of the release of radioactive materials.

To this end the safety systems ensuring the realisation of the safety functions have to be protected against fire effects so that the necessary number of redundancies remains available for every postulated fire.

For the newer nuclear power plants, detailed requirements with regard to the fire protection concept and the measures to be taken are laid down in KTA safety standard 2101. However, this safety standard was developed after the construction of the first German NPPs in accordance with the worldwide progress of the safety culture and the fire protection requirements for nuclear power plants.

Three “generations” of NPPs can be distinguished in Germany with regard to the fire protection concept at their start of operation. The fire protection measures of the first generation of NPPs were designed according to general fire safety requirements for buildings and industrial plants. They were supervised by the licensing authorities for building construction and, in general, were not subject to requirements derived from nuclear safety.

The second plant generation was under construction when the significance of the fire protection measures for the nuclear safety was recognized. Therefore, the fire protection concept was reviewed by the licensing authorities for nuclear safety during the installation. This resulted in substantial improvements of the fire protection concept before completion, whereas at the oldest plants upgrading started after the beginning of commercial operation.
Finally, the third generation of plants was designed according to safety standard KTA 2101.

The three generations of fire protection concepts of German NPPs at completion are characterized by the following features:

- **1st Plant generation:**
  - design according to conventional fire protection requirements,
  - no physical separation of safety systems' redundancies,
  - high importance of fire extinguishing.

- **2nd Plant generation:**
  - partial physical separation of safety systems' redundancies,
  - fire barriers for areas or rooms with high fire loads,
  - stationary fire extinguishing systems in areas with safety related systems or higher fire loads.

- **3rd Plant generation:**
  - design according to KTA safety standards,
  - physical separation of safety systems' redundancies,
  - all fire barriers with a fire resistance rating of 90 minutes (F 90),
  - stationary fire extinguishing systems in compartments with high fire loads.

These characteristics apply to the plants as built. Since then, the fire protection measures as well as the safety systems of the first and second generation plants have been upgraded. The analysis showed that the independent emergency systems (IES), installed in all plants for external event conditions, also have advantages for fire safety.

As a result buildings with different characteristics for the structural fire protection measures can be found in plants of the 1st and the 2nd generation (figure 1):
- The main buildings are not designed or partially designed for a structural separation of redundant safety systems. Inside the building, an upgrading of structural measures is done as much as convenient. In addition, a variety of fire extinguishing systems was installed.

- In the buildings for the independent emergency systems, redundant trains are separated physically. The fire protection in these buildings is designed on the basis of the KTA safety standard.

In consequence, a fire protection concept with two main aspects has been developed:

- In the older parts of the plants fire protection is based on upgraded passive, active and administrative fire protection measures with the following goals:
  - effective prevention of the ignition of fire loads,
  - fire fighting during the pre-flashover phase, where possible,
  - confinement of a fire in the building, in which the fire started,
  - prevention of the fire spreading to adjacent buildings.

- Independent emergency systems in separate buildings provide for the functions lost due to a fire damaging more than one redundancy. In this context, the importance of the IES in case of large fires is more considerable for the 1st plant generation.

3 Results of plant specific reviews

The GRS participated in several reviews and was concerned with the demands on the fire protection systems regarding the nuclear safety. The most important results of some analyses concerning separation of fire compartments, fire detection, and fire extinguishing systems are presented.

3.1 Separation of fire compartments

According to the general regulations by the building supervising authorities, fire compartments in NPPs have to be separated by walls and structures in F90 quality. For
new plants this requirement is applied everywhere, even in those areas where fire spreading seems impossible, e.g. due to the absence of fire loads.

Figure 2 shows an example for the physical separation of safety redundancies in the switchgear building for a plant of the 3rd generation (new plant). The building is divided into four parts for each of the redundant equipment of the emergency power supply, the reactor protection system and the instrumentation and control systems. Each part of the building is consequently separated from the other parts and has its own independent ventilation system. Also the different floors of the building are separated by fire barriers.

In comparison to figure 2, figure 3 shows an example for the physical separation of safety systems redundancies in the switchgear building for a plant of the 2nd generation. The separation of the redundancies was realized during construction of the building. Some backfitting of the fire barriers was necessary as a result of the fire safety review.

There is one common ventilation system for the building. For that reason, lots of fire dampers are installed in the ventilation openings of the fire compartments. Similarly, in those locations where ventilation channels lead through a fire compartment to a room of an other system redundancy, those channels are designed with a fire resistance rating of 90 minutes.

As a third example, figure 4 shows a simplified plan of a site of an older German PWR of the 1st plant generation. It was built in the late 1960ies. During the whole plant life the safety systems have been improved according to the technical development. Among others, separated emergency systems have been added, and the fire protection system has been upgraded by various measures. Now, the fire protection system follows the concept outlined in the previous chapter, that means, it consists of original buildings with upgraded fire protection systems and of separated independent emergency safety systems (IES). These now constitute a significant part of the fire protection concept. As a result of the review, the fire safety level of this plant does not essentially differ from the safety level of newer plants.
3.2 Fire detection and stationary fire extinguishing systems

Fire detection systems are installed in all buildings and areas relevant for the plant safety. The detector systems installed with regard to the upgrading measures combine different types of detectors: Heat, flame, smoke in, and particular, ionisation detectors, or other optical detection systems to achieve the present state of the art.

Mainly, a static detection process via detection lines is used for the detection systems. In some plants pulse signal detection systems are installed. Computer assisted fire detection systems can be found in many German NPPs. The plant specific number of detectors is varying from 1000 to 5000. For all plant areas fire alarms are signaled in the alarm receiving station in the main control room and, in some cases, also in the emergency control room for those areas affected in case of emergency. In some German plants individual fire alarm panels can be found for all buildings.

With respect to manual fire fighting the results of the reviews showed that there is a sufficient number of mobile fire extinguishers existing in all plant areas. But differences were found in the location of the extinguishing water supply systems as well as differences with respect to personal and technical equipment of the fire brigades. Backfitting measures concerning the water supply systems and the fire brigades have already be done or are intended to be done.

For stationary fire extinguishing systems the results of the reviews showed plant specific deficiencies with regard to location and effectivity for some NPPs. The consequences are:

- In many plants additional fire extinguishing measures have been installed and
- already existing systems have been replaced by more efficient ones to provide for efficient fire fighting in hardly accessible areas and to prevent the spreading of fires to other redundancies.

However, the efficiency of fire suppression depends on the early actuation, particularly, of the stationary fire extinguishing systems.

Concerning the actuation of such systems, the reviews gave the following results:

- An automatic actuation is realized only for a few selected areas.
Mainly, a manual actuation is realized with

- an actuation in locations adjacent to the fire compartments,
- an actuation from the main control room, for a lot of systems, but not for all.

In case of manual actuation, an essentially weak point is the significant time delay between fire alarm, fire verification and actuation of the fire extinguishing systems. That is a weak point in general basing on the concept to verify the fire by a walk down of the plant personnel to the fire compartment.

As a consequence, it is recommended to check the opportunities of a faster manual or automatic actuation of stationary fire extinguishing systems. In every single case one should consider:

- the safety relevance of the protected area,
- the conditions for the fire spreading,
- the effects of spurious actuation,
- the requirements for a fast manual actuation.

Where necessary, the actuation method should be improved. Moreover exactly defined criteria for the actuation of these systems should be established. In this context, the question of manual or automatic actuation is of special importance. It will be discussed in the next chapter.

4 Consideration of manual and automatic actuation

In the first part of this chapter the principle advantages and disadvantages of the automatic actuation of fire extinguishing systems are discussed. Experiences with fires and fire fighting systems in nuclear power plants are analysed, basing on national and international fire incidents. Finally, the current upgrading practice for German NPPs is presented.
4.1 Advantages and disadvantages of automatic actuation

The main advantage of an automatic actuation of the fire extinguishing systems is the limitation of the maximum fire size and of the resulting damage, as fire fighting starts early during the ignition phase. Therefore the probability of a successful extinguishing is high. The success mainly depends on the reliability and the effectivity of the technical system and the amount of extinguishing agent in relation to the fire size. In contrast, at delayed manual actuation the fire extinguishing system might be badly damaged even a short time after the beginning of the fire, making an efficient fire fighting impossible. If fire fighting does not start before the fire is fully developed the amount of extinguishing agent might not be sufficient to put the fire out.

An automatically actuated systems limits the damage caused by the fire due to the early start of fire fighting, the development of smoke as well as the release of toxic and corrosive gases. Only the systems and components close to the fire source are affected. A fire spreading to adjacent rooms or areas is prevented.

The disadvantage of automatic actuation is the danger of an spurious actuation of the fire extinguishing system. This may not only cause expensive damage to equipment, but may also affect the nuclear safety (temperature shock, water sensitivity). In controlled areas additional problems have to be faced when fire water is contaminated and consequently has to be held back and treated as radioactive waste.

Initiators for spurious actuation can be technical defects of the technical systems itself, of the actuation mechanism or of the fire detection system as well as spurious alarms caused by deceptive circumstances (smoke, dust, steam, heat). According to /DÄT 84/ the latter is the main cause for spurious signals, being responsible for about 80 to 90 % of all spurious signals. The ratio of spurious to real alarms amounts to 14:1 according to this reference. This number compares well with the results given by other authors.

The frequencies of spurious actuation are significantly influenced by the actuation criterion, i.e. actuation by 1 of 1 alarm signal of a detection line. In general, for NPPs 2 of 2 actuation criteria are consideral. Therefore the frequencies for spurious actuation are lower than in systems installed in conventional buildings. But for the present state of the art of actuation technics spurious actuation cannot be excluded.
4.2 Incidents involving fires and fire fighting systems

4.2.1 International experience

In an analysis of international incidents, Ishak found 57 incidents involving fires and/or fire extinguishing systems out of approximately 1400 incident descriptions in the NEA-IRS data base /ISH 90/, where only incidents relevant for the nuclear safety are reported. Out of these 57 incidents only 34 were fire events, whereas 23 incidents involved fire extinguishing systems without any fire. However, not in all of the latter cases a fire extinguishing system was actuated with resulting damage to equipment. This applies only to a few incidents:

- In four cases a deluge system was inadvertently actuated due to human errors.

- For two incidents spurious actuations of automatic fire extinguishing systems occurred (in 1 case due to high humidity, in the other due to a steam leakage increasing the room temperature).

- In three cases deluge systems were actuated by smoke or ionization detectors because of smoke caused by other sources than fires (welding, hot bearing, hot motor).

- Four incidents comprised actuations of fire suppression systems due to design peculiarities or other errors not related to the actuation method.

In conclusion, only 5 incidents were reported to the NEA-IRS data base for which an automatic fire extinguishing system was actuated inadvertently by the fire detection system resulting in substantial damage and/or non-availability of safety systems. The other incidents mentioned were not dependent on automatic actuation.

Of course, these are not all incidents with spurious actuation that ever occurred in nuclear power plants, but only those with relevance to the nuclear safety, which were consequently reported the NEA-IRS data base. But that may be an indicator for the significance of the actuation method for the fire extinguishing systems.
4.2.2 German experience

In the German nuclear incident file more than 3700 incidents have been recorded up to the end of 1992. In a recent survey 41 incidents involving fires and fire detection systems were identified. 22 of these incidents were actual fire events, whereas in 19 cases problems with fire protection systems were found. Statistical data concerning spurious actuations are not available, but it is well known that spurious actuations also occur in German NPPs. Up to now, no safety significant incidents with regard to spurious actuations of fire extinguishing systems are reported. The causes for the occurrence of spurious actuations are comparable with the international experiences. For example, the generation of smoke by other sources than fires, e.g.

- short circuits or other electrical failures,
- overheating of bearings,
- leakage of exhaust fumes into the Diesel engine room

may be typical for the spurious actuation.

4.3 Current actuation practice in German nuclear power plants

According to safety standard KTA 2101.1, stationary fire extinguishing systems basically should have an automatic actuation mechanism. Manual actuation is allowed if the fire effects can be controlled until effective fire fighting starts. On deciding upon the actuation method, the impact of an inadvertent actuation of the fire extinguishing system on safety related equipment has to be taken into account.

This safety standard has to be applied for new nuclear power plants. It is used as a guideline when upgrading the fire protection systems of older plants. Consequently, the situation has to be analysed specifically for each system. This should be demonstrated with an example in the following.

In one reference plant only manual actuation schemes could be found during the first step of review:

- Some fire extinguishing systems are started manually from the main control room or with the alternative of manual actuation at the system. Examples for these are
the water deluge systems in cable ducts and cellars and the CO₂-systems in the Diesel fuel storage rooms and for the oil supply system in the turbine building.

- Water deluge systems in some areas of safety significance can be actuated manually from the emergency control room or inside the extra building for the deluge systems.

The final review of the plant had the result that 5 of 12 fire extinguishing systems actuated manually will be changed to automatic actuation. Manual actuation was preferred in the following cases:

- Oil supply of the main coolant pumps. The fire load is enclosed by fire barriers. Early fire detection is guaranteed by smoke detectors. A fast fire verification is realized by two detector lines with different types of detectors and a video camera system in this area. Fast growing of the fire is prevented by interruption of the fresh air supply with automatic fire dampers. Therefore, efficient and fast fire fighting is possible with manual actuation of the stationary water-foam system.

- Diesel engines for safety injection pumps and emergency power system. For Diesel engine rooms the probability of a spurious fire alarm is much higher when the engine is running than at standstill. It would be harmful if the inadvertent actuation of a fire extinguishing system caused the failure of a running Diesel just when it is needed. Generally, the availability of Diesel driven safety systems is more important than the gaining of time for fire fighting by automatic actuation, provided a fast fire detection is guaranteed.

- In cases where an early fire detection and verification is certain, for example in the computer room. The CO₂-system can be actuated fast enough to provide for efficient fire fighting. The effects of inadvertent actuation on persons or equipment could be problematic.

On the other hand, automatic actuation was preferred because of the following reasons:

- difficult accessibility for manual fire fighting so that a failure of the stationary fire extinguishing systems caused by too late actuation has to be prevented,
- no danger for safety related equipment or persons due to spurious actuation,
minimization of the risk of spurious alarms by sophisticated choice and location of fire detectors and 2 of 2 actuation criteria.

Examples, where automatic actuation was adopted, are the CO₂-systems in cable spreading rooms, which are usually not occupied by personnel, and water deluge systems (4 sections) protecting cable trays in the annulus of the reactor building.

5 Conclusions

Depending on the generation of the fire protection development during construction of German NPPs, important differences in extension and quality of separation of safety systems redundancies have been found. Due to the real building layout and the impossible change of plant conditions, an elimination of deficiencies in structural fire protection measures can often not be realized. For plants of these generations, the following concept for fire protection has to be considered:

- Backfitting of structural measures as much as convenient,
- Additional fire extinguishing systems,
- Technical support by the independent bunkerized emergency system (IES) in case of damage of more than one redundancy of a safety system being necessary for that special situation. The IES and its connections to the reactor building should be designed against those simultaneous failures in a way which surely exclude fire effects.

The results of the reviews do not show important deficiencies of the fire detection systems in the plants. Concerning the fire extinguishing systems, recommendations for replacement of systems or additional systems have been given. In particular, the water supply systems for fire fighting have been completed in some NPPs.

As a consequence of the results of the review the time delay between a fire alarm and the actuation of the fire fighting has been eliminated as far as possible. This could be realized by a far-reaching change from manual actuation to automatic actuation of stationary fire extinguishing systems. However, manual actuation is still preferred, also in future, if important disadvantages, in particular for the plant safety, cannot be
excluded. In these cases special requirements for a fast detection and verification of a fire are recommended to exclude a significant time delay.

6 References

/DÄT 84/ Dätwiler, F.; Preissard, W. G.: "Fehlalarm"; Neue Züricher Zeitung, 16.05.1984, Nr. 113, S. 65

Figure 1: Upgrading Structural Fire Protection for the three Plant Generations
Figure 2: Example for the Physical Separation of the Safety Systems Redundancies in the Switchgear building for the 3rd Plant Generation
switchgear building (level: 4 m - 8 m)

turbine building

reactor building

Unit systems
Redundancy 1
Redundancy 2
Redundancy 3

Figure 3: Example for the Physical Separation of the Safety Systems Redundancies in the Switchgear building for the 2nd Plant Generation
Figure 4: Example for the Physical Separation of the Safety Systems Redundancies in the Switchgear building for the 1st Plant Generation
Fire Protection Management in
Japanese Nuclear Power Plants

1. Introduction

Fires in Japanese nuclear power plant is very seldom event. Only four events were reportable in the past experience of operation for over 27 years. (Ref. Table 1)

In 1980, Nuclear Safety Commission has issued "The Review Guide for the Fire Protection of the Light Water Type Power Generation Nuclear Reactor Facilities" which requires the utility companies necessary items for fire protection measures to be included in their safety analysis report in the application for establishment permission of the nuclear power plants (Ref. Appendix 1)

However, electric utilities have operated their thermal power plants for long time before the first introduction of nuclear power plant.

Thermal power plants have storages of large quantity of fuel, either coal, or oil, which are the subject of the Fire Protection Law. Therefore, fire protection management was major concern of the utility companies and they have established administration guidelines based on the legal requirement.


Usually, as practices on fire protection management are different for individual nuclear power stations, following information is an example of the specific nuclear power station:

2.1 Organization and routines for daily fire protection management

The organization of fire protection management for the nuclear power station are consisted by:
(1) Fire Protection Management Committee

The committee is consisted by the General Manager of the station as the Chairman, the Deputy General Manager as the Vice-Chairman and Managers of the individual organizations and the Fire Protection Administrator of the station.

This committee reviews:
- Modification of fire protection plan of the station
- Maintenance / management of the fire protection facilities of the station
- Organization of self defense fire brigade
- Education and training on fire protection and
- Other matters

Committee is held twice a year or anytime requested by the chairman as necessary.

(2) Fire Protection Management Organization

Deputy Fire Protection Administrators are nominated to support the Administrator and Responsible Person in charge is nominated under the Responsible Person for Fire Protection who is responsible for individual subjective items for fire protection. Within the protection zone, including reactor building, auxiliary building, turbine building, etc, fire protection is generally administrated by the chief supervisor for operation. Outside of the protection zone including main administration building, general information building, modification work office, main building of the power station, etc, are devided by room base classification and responsible adminiscurators or responsible persons are defined. Also, detailed inspection list for facilities is defined.

(3) Self Defense Fire Brigade System

Self Defense Fire Brigades are organized for two zones

(1) Within the protection zone

The response headquarter headed by the General Manager of the station and the local direction center headed by the Deputy General Manager are the main organization.
(ii) Outside of the protection zone
   The same system as above is organized.

   Detailed organizations of the Fire Brigades for two zones are also defined in details. Typical organization chart of a fire brigade is shown in Fig.1.

2.2 Routine check practice for fire protection
   From the standpoint of safety protection, safety securing and fire protection managements, routine patrol and checking are executed as follows:

(1) Within the protection zone
   (i) Reactor facility--Staffs of the shift are required to inspect following facilities and equipment along the predefined patrol course more than one time a day.
      (a) Reactor cooling system facility
      (b) Control rod driving equipment
      (c) Power supply, feed and discharge water and waste facilities
   (ii) Service building--Guard staffs patrol two times a day

(2) Outside of the protection zone
   (i) Main administration building--Guard staff patrols to check individual rooms of the building nine (9) times a day
   (ii) Area around the surrounding watch fences within the power station ground is checked by the guard nine (9) times a day
   (iii) Area around the surrounding watch fences outside of the power station ground is checked by the guard nine (9) times a day

2.3 Reporting route and response actions of the fire
   In case of fire occurrence, reporting routes are defined precisely. Some examples for reporting route and response actions are shown in Fig.2 and Fig.3.
3. Conclusion

As mentioned before, fire event reportable to the regulatory body was only four cases in the past operating experience for 27 years.

Reason for such fireless situation may be attributed as follows:

1. Very severe requirement and administration for fire protection defined by law and executive regulations.

2. Strict check of possibility of fire by well equipped detection system and routine administration of the utility themselves.

3. High level of education for utility staffs and positive intention of preventing fire by them.

Fire protection is solely responsible of the site staffs of utility and integration of the well maintained equipment, consciousness and positive intention of the staffs of the power station, combined with the severe legal requirement and strict administration are giving actual good results.
<table>
<thead>
<tr>
<th>Name of Utility</th>
<th>Date of Occurrence</th>
<th>Name of Plant</th>
<th>Event Title</th>
</tr>
</thead>
<tbody>
<tr>
<td>JAPC</td>
<td>Nov.18,1967</td>
<td>Tokai</td>
<td>Mortality event due to the fire in the No.2 gas circulator room (L)</td>
</tr>
<tr>
<td>TEPCO</td>
<td>Mar.25,1977</td>
<td>Fukushima Daiich-1</td>
<td>Trouble occurred during fusion cutting work in the reactor building (N)</td>
</tr>
<tr>
<td>TEPCO</td>
<td>Aug.31,1985</td>
<td>Fukushima Daiich-1</td>
<td>Burnt down of the start up power bus cubicle (N)</td>
</tr>
<tr>
<td>TEPCO</td>
<td>Jan.13,1988</td>
<td>Fukushima Daiich-6</td>
<td>Small fire in the feed air fan room of the turbine building (N)</td>
</tr>
</tbody>
</table>

Note (L) event reportable under the provisions of the law.
(N) event reportable under the provisions of the notification of MITI
Review Guide for the Fire Protection of the
Light Water Type Power Generation Nuclear Reactor Facilities

November 6, 1980
(Partially revised on August 30, 1990)

I  Forward
II  Definition of the Technical Terms
III Review Guide on the Fire Protection

For the prevention of the degradation on the safety of nuclear reactor facilities by fire, design shall be such that appropriate combination of following three measures: prevention of fire occurrence, detection of fire and mitigation of effect of fire according to the importance of safety functions.

1. Prevention of fire occurrence

1-1 Design of the reactor facility shall have the preventive measures for the occurrence of fire not only during its normal operation but also in the abnormal conditions.

1-2 For structures, systems and components having safety functions, design shall apply non-burnable or flame retardative materials as much as practicable.

1-3 For structures, systems and components within the reactor facilities, design shall protect them from the occurrence of fire due to the natural phenomena such as thunderbolt or earthquake.

2. Detection and extinguishing of fire

2-1 Detecting and extinguishing facilities of fire shall be designed to limit the non-preferable effects of fire of the structures, systems and components having safety functions and to be able to distinguish the fire in its early stage.
2-2 Fire extinguishing facility shall be designed not to degrade the safety functions of the structures, systems and components by its failure, malfunction or misoperation.

2-3 Fire extinguishing facility shall be designed not to degrade greatly its functions by the natural phenomena occurred in association with the fire.

3. Mitigation of the effect of fire

3-1 The area including the structures, systems and components having safety functions, shall have the mitigation measure of fire effect according to its importance including the effect of fire of the adjacent areas.

3-2 For postulated fire occurred in any location of the reactor facility, the design shall be such that in the case that the disturbance reaches reactor itself and actuation of the safety protection and reactor shutdown systems are required, reactor shall be able to hot shutdown even in single failure condition.

The systems necessary for cold shutdown shall be designed not to degrade their functions by the fire postulated at any location within the reactor facility.
Fig 1 Organization Chart of Self Defense Fire Brigade

Commander of the Self Defence Fire Brigade
- Control of the activities of the Self Defense Fire Brigade and grasping Overall Situation
- Provide information to the Fire Brigade
- Instruction to the personnel within the station buildings
- Communication to the related bodies and / or personnel
- Operation of fire extinguishing equipment
- Grasping of evacuation conditions

Reporting and Communication Group
- Reporting to the fire protection organizations and designated places as administration positions, etc.
- Notification to the personnel within the station building
- Communication to the adjacent rooms and / or buildings

Fire Extinguishing Group
- Fire extinguishing activities using fire extinguisher and / or hydrant
- Prevention of loss of water storage by the fire extinguishing activities

Evacuation Guiding Group
- Guide for the personnel within the station building and confirmation of number of personnel
- Opening of the emergency exit and closing of openings after evacuation
- Set up of the equipment for evacuation
- Guide to the evacuation place

Safety Protection Group
- Operation of fire proofing doors and / or shutters
- Operation of fireproofing damper
- Operation of smoke-exhausting equipment
- Emergency response measures for dangerous objects gas, and / or fire handling equipment

Rescue and First-aid Group
- Carrying out of wounded personnel
- First-aid measures for wounded personnel

Emergency Carry-out Group
- Carrying out and management of important documents
Fig. 2 Reporting / Communication Chart for the Fire Occurrence in the Night of Ordinary Working Days

- General Manager of the Power Station
- Deputy General Manager of the Power Station

- Director of Power Generation Dept
- Director of Administration Dept

- Director of Public Information Dept
- Director of Engineering Dept

- shift supervisor

- Fire Station
- Fire Protection Administrator

- Safety Group
- Manager of Administration Sect
- Manager of Labour Sect
- Manager of Protection Management Sect

- Manager of Engineering Sect.
- Manager of Safety Securing Sect
- Chief of Health and Safety Center

- Nuclear Power Operating Administration Office, MITI
- Manager of Nuclear Power Generation Section (Headquarters)
- Regional Office of MITI
- Stationed Senior Operating Specialist (MITI)
Fig. 3 Response Action Flow Chart for Fire Occurrence (Day / Night Time of Ordinary Working Days)

Occurrence of Fire

First Finder
Report to the Shift Supervisor

Shift Supervisor
Direct immediate early stage fire extinguishing activity by the deputy supervisor and other staffs

Actuation of Fire Alarm

Shift Supervisor
Direct immediate confirmation of fire and place by the deputy supervisor and other staffs

Deputy Supervisor and Other Staffs
Confirmation of fire

Early stage extinguishing activities

Deputy Supervisor
Report to the Shift Supervisor

Shift Supervisor
Reporting / Communication

Fire Station (119)

Fire Protection Administrator
Director of Power Generation Dept
ANNUAL REPORT ON FIRE PROTECTION AT RINGHALS NUCLEAR POWER STATION.

PICTURE 1

My name is Tommy Magnusson and I work as fire protection engineer at Ringhals... I started 1976 at Ringhals as a health physics technician... 1979 I started to work with fire protection. 1983 I was graduated as fire protection engineer.
Apart from fire protection I also work with emergency preparedness.
I'm responsible for every part of the work that is named in the report. Service testing of fire equipment as well as the fire brigade and all fire exercises.
I'm also responsible for all connections with the authorities concerning fire protection.
Ringhals consists of four plants.
Ringhals unit 1 has an electrical output of 795 MW. The plant is constructed by ASEA ATOM. The plant was taken in operation 1976.
Ringhals 2, 3 and 4 are Westinghouse construction and were taken in operation respectively 1975 1981 and 1983.
I work in the line organisation I also have a staff mission concerning everything that has to do with the fire protection at the Ringhals units. This includes finding the right level of the fire protection.
Also recommending different policies about fire protection and make instructions for the organisation.
The organisation of Ringhals consists of two profit centres. Unit 1 and 2 and unit 3 and 4.
Each unit has its own responsibility as to follow different rules and regulations.
Ringhals 3 and 4 are no big problems.
Ringhals 1 and 2 are an early generation and the fire safety systems is of course of on old construction.
therefor it shouldn't be possible to have the same demands at the four units. But unfortunately in practice that is just what happens. The only exception from this that differs between the four units is the demand upon the basic construction.

PICTURE 2
REPLACED WITH
MAP OVER LOCATION OF RINGHALS NUCLEAR PLANT

PICTURE 3

I have since 1984 summarised an annual report on fire protection.
The report has had different kinds of contents.
The following is the contents of the report and for the last three years it has been made in the same way.
Reports on fire protection normally consists only of turnouts and maybe also of fire inspections which have been made.

PICTURE 4

Although I make all tests and service of the fire equipment it is the manager of each plant who has to gather all facts concerning their own unit to the report.
When I we got all facts i have to make a judgements of all these received facts.
And or instance see what precautions to make out of each evaluated fact.
What precautions we have to make when the fact is evaluated.
It's important that the report is signed by each plant manager and the director of Ringhals. The report is distributed to local and central fire authorities, nuclear inspection authority insurance company and other nuclear plants in Sweden.

I'm sorry but I haven't had time to translate one whole report into English but I will do so in due time.

In the report I recommend some actions to be done based upon the results of the entire report. This can be a technical advice or advice about improvements of the organisation.

**Picture 5**

In a written agreement we have decided what obligations Ringhals and the local fire authorities have towards each other.

Once a year the local fire authorities make an inspection at each plant. The plant manager has the full responsibility to take proper actions regarding the remarks wish has been noted under the inspection.

At Ringhals we have a full time rescue force. An officer in command who also is an health physics engineer. We also have four fireman and they normally work as security guards. The fire officer in command at Ringhals is also in charge of the local part time fire brigade. In Varberg the nearest town we have the local fire authorities. In case of fire at Ringhals we are in charge until the local officer in command takes over the control in which takes in about 25 minutes.

We try to exchange experience between other local fire authorities and other Swedish nuclear plants.

Every year we have at least three big fire drills and a lot of minor training events. In case of incidents or other events which can lead to a fire, Ringhals reports to the local fire authorities.

**Picture 6-12**

No written comments.

**Picture 13**

This is an example of testing results covering a period of three years.

The test is done once a year.

In the diagram you can see the number of remarks.

In the margin you can see how many doors we have in each unit.

The remarks are based upon problems with door closers, how tight they fit, signs, if they are 60 minutes fire proof, and the locks work properly.

The trend is raising in the beginning but after proper service between tests the number of remarks are falling in the end.

**Picture 14**

During service of for instance a water sprinkler system we have to make special precautions while the system is out of order. The technical exchange is normally same extra fireman with special equipment. How long time, which system has been taken out of order and which precautions have been made.

If other Nuclear Plants would make the similar report it would be possible to compare what acceptable level is.

**Picture 15**

No written comment.
ANNUAL REPORT ON FIRE PROTECTION
AT RINGHALS NUCLEAR POWER PLANT

UNIT 1  BWR 795 MW ASEA ATOM  1976
UNIT 2  PWR 875 MW WESTINGHOUSE 1975
UNIT 3  PWR 915 MW WESTINGHOUSE 1981
UNIT 4  PWR 915 MW WESTINGHOUSE 1983

TOMMY MAGNUSSON
FIRE PROTECTION ENGINEER
DEMANDS

REACTOR SAFETY  NUCLEAR SAFETY AUTHORITY

PERSONNEL SAFETY  INDUSTRIAL WELFARE AUTHORITY

ECONOMIC SAFETY  INSURANCE COMPANY
PURPOSE WITH REPORT ON FIRE PROTECTION

TO GIVE A GENERAL VIEW OF THE WORK WITH FIRE PROTECTION AT RINGHALS

TO PRESENT THE RELIABILITY OF FIRE EQUIPMENT

TO PRESENT WHICH IMPROVEMENTS THAT ARE NEEDED IN TESTING, MAINTENANCE PROCEDURES AND EDUCATION

TO COMPARE THE TRENDS AND THE RESULTS FROM YEAR TO YEAR

TO SHOW WHAT THE IMPROVEMENT ARE
REPORT IN GENERAL

GATHER FACTS OF DIFFERENT KINDS

JUDGEMENT OF FACTS

CONTROL/SIGNATURE

DISTRIBUTION

ACTIONS
CO-OPERATION WITH THE LOCAL FIRE AUTHORITIES

INSPECTION BY THE LOCAL FIRE AUTHORITIES

RINGHALS RESCUE FORCE

OFFICER IN COMMAND

EXCHANGE OF EXPERIENCE

EXERCISES TOGETHER WITH THE LOCAL FIRE BRIGADE

INFORMATION REGARDING DIFFERENT EVENTS AT RINGHALS
THE REPORT

FIRE INSPECTIONS

FIRE INSPECTIONS BY THE LOCAL FIRE AUTHORITIES

EVERY THREE ANNUAL INSPECTIONS

INSPECTION AFTER OUTAGE

INSPECTION UNDER OUTAGE

RISK ANALYSIS

(INSURANCE COMPANY)
REPORTS TO NUCLEAR INSPECTION AUTHORITIES

EXAMPLE/ FIRE DIESEL PUMP OUT OF ORDER
TESTING

NUMBER OF COMPONENTS IN APPENDIX

FIRE PUMPS

SPRINKLER VALVES

NOZZLES AND PIPES OF THE WATER SPRINKLER SYSTEM

SMOKE DETECTORS

ALARM BUTTONS

 SIGNALS FROM THE FIRE ALARM PANELS TO FIRE PUMPS, FIREDAMP'S, FIREDOORS ETC.

HALON EXTINGUISHING SYSTEM

CARBON DIOXIDE EXTINGUISHING SYSTEM

SMOKE VENTILATORS

THERMAL FIREDAMP'S

FIRE EXTINGUISHERS

INDOOR HYDRANTS

OUT DOOR HYDRANTS
STATISTICS FROM THE FIRE ALARM PANEL

TOTAL NUMBER OF ALARMS IS ABOUT 150/YEAR

10 - 25 / UNIT

98% ARE NOT CAUSED BY FIRE

140 TECHNICAL DISTURBANCES

1100 TECHNICAL MEASURES TO AVOID FALSE ALARMS

200 CHANGES TO HEAT DETECTORS

THE FIRE ALARMS CAN BE ACTIVATED FOR SEVERAL REASONS SUCH AS WELDING AND DRILLING
STATISTICS FROM THE PLANT FIRE BRIGADE

NUMBER OF TURN OUTS

ABOUT 110 - 130 / YEAR

NUMBER OF RESCUE OPERATIONS

ABOUT 5 / YEAR

RESCUE OPERATIONS

FIRE

RESCUE OF LIFE

OIL OUTLET
**FIRE TRAINING**

**TYPE OF TRAINING**

**TYPE OF PERSONNEL**

**NUMBER OF HOURS**

**TOTAL AMOUNT OF TRAINING IS ABOUT 10000 HOURS / YEAR (1100 EMPLOYEE)**

<table>
<thead>
<tr>
<th></th>
<th>1/3</th>
<th>100  H/Y</th>
</tr>
</thead>
<tbody>
<tr>
<td>FIRE BRIGADE</td>
<td></td>
<td></td>
</tr>
<tr>
<td>OPERATIONAL PERSONNEL</td>
<td>1/3</td>
<td>12 H/Y</td>
</tr>
<tr>
<td>REST</td>
<td>1/3</td>
<td>4 H/3Y</td>
</tr>
</tbody>
</table>
IMPROVEMENTS ON FIRE PREVENTION

EXAMPLE

UNIT 1 / 91

MARKING OF EVACUATION ROUTES ON THE FLOOR 5000 M

SPRINKLER INSTALLED IN OIL ROOM

INCREASE OF FIRE DETECTORS IN DIESEL ROOM
EXAMPLE
TESTING FIRE DOORS
ONCE A YEAR
REMARKS

DOORS

UNIT 1 303
UNIT 2 300
UNIT 4 303
UNIT 3 316

DOOR CLOSER
FIT TIGHT
SIGN, FIRE DOOR
FIRE PROOF 60 MINUTES
LOCK

253
DEVELOPMENT

TECHNICAL EXCHANGE

THE TIME THE FIRE BRIGADE NOT IS AVAILABLE

SHOW THE TRENDS FOR EACH SUBJECT

COMPARE THE TRENDS AND THE DEVELOPMENT FOR EACH SUBJECT
A GOOD FIRE PROTECTION IS NOT STRONGER THAN THE WEAKEST LINK

THE REPORT SHALL GIVE INFORMATION ABOUT ALL ASPECTS OF THE FIRE PREVENTION

EVERY WEAKNESS SHOULD BE SHOWN IN THE REPORT AND LEAD TO APPROPRIATE ACTIONS
Session #4

Research and Development

Mr. Maki, Chairman
FREIA

A Risk Analysis Tool for Accidental Releases and Fires in Industrial Buildings

presented by

Fredrik Jörud
Fire Protection Engineer

SYDKRAFT KONSULT AB
S-205 09 MALMÖ
Sweden
Introduction

FREIA, the acronym for FiRe Explosion Industrial Assessment, is an "expert" system used as a risk analysis tool for accidental releases and fires in industrial buildings and power-plant facilities. The program can also be used in other operating areas such as the process industry.

FREIA has been developed as a cooperative effort between the department of Fire Safety Engineering Institute of Science and Technology, Lund University, Sweden and Sydkraft Konsult AB, Sweden. The work began in late 1989 and was concluded in June 1993.

The program's application areas

FREIA deals with fires in conventional solid material and the accidental release of toxic and/or combustible gases and fluids such as natural gas, propane, chlorine, ammonia, heating oil, etc.

The system calculates the consequences for both humans and components in the vicinity of the fire and/or the release. Included in the system are effects of active systems such as sprinkler suppression, explosion relief vents, smoke ventilation, gas and smoke detectors and carbon dioxide gasphase extinguishment. Humans and components can be affected by heat, pressure and toxic gases. For scenarios inside a building the result of an evaluation is given as estimated time to reach component failure criteria. The geometric positions of the components in the room is considered but not shadow effects. For scenarios affecting areas outside the building the consequences for human beings are considered to be of primary interest. But even so it is possible, if the user prefers, to get the result for a specified component. However, a substantial lack of data when components are damaged has been revealed and therefore it is difficult to estimate the exact time when a component is damaged.

The term "expert system"

The description "expert system" defies uniform definition. It is the accepted collective designation of a type of computer program which, based on human-knowledge input within a specific area of know-how, is capable of a form of "intelligent" problem-solving.

Typical characteristics of an expert system are included in the following definition:

* The system is based on expert knowledge within a particular application area. The knowledge may pertain to quantitative or qualitative facts but also to rules of reasoning, including reasoning under uncertain conditions and intuitive knowledge which are difficult to formulate unequivocally.

* In the system, the realization of expert knowledge is through a gradual build-up of a knowledge database. This knowledge database contains facts and rules that form the basis for decision-making.
An example of the FREIA expert system would be to issue advice to the user, in situations where it is impossible to compute a detailed scenario. For instance, gas release impacts a wall or floor, making it impossible to calculate the concentration field near the release. The user then obtains an expert judgement of the concentration at this location.

Advantages compared with other programs
Compared with other computerized calculation programs which assess fire and accidental release consequences, FREIA has the advantage:

* It considers both conventional fires such as fires in cable trays and accidental releases. It also takes into consideration the interaction between fires and accidental releases. For instance, an initial fire might result in an accidental release.

* Releases inside and outside a building are considered.

* It is an expert system. That means that even if it is not possible to calculate consequences in detail, advice is given to the user.

System build-up
The structure of the system is shown in figure 1. As seen in the figure, the system contains a preprocessor for producing input data to the system. This preprocessor includes databases for:

* building materials fire response
* physical data of gases and liquids
* damage criteria for components
* a statistical data on accidental releases and ignition of these

All databases can be revised and updated by the user.

The pre-processor also contains tools for generating space-comprehensive input-data files. These may include type of surrounding structures, presence of sprinklers and fire alarms, openings, mechanical ventilation, etc.

The evaluation part of the system is in principle divided into two subdivisions, one fire part and one accidental release part. However, these two parts are connected. Each part is built on well known validated models. When starting an evaluation the user first has to decide whether to consider a fire or an accidental release as initial event. Some help choosing initial scenario is given by the statistical database.

For the development of the system a commercial expert system shell named Nexpert Object was used. Incorporated with the knowledge base developed with the aid of Nexpert Object are several computer programs which calculate concentration fields, radiation, temperatures, etc. Integrated with the system is for example a program for calculation of dispersion of dense gas clouds.
Figure 1: System build-up
Accidental releases

The release scenarios are continuous and instantaneous releases of toxic and/or combustible gases and liquids. Both pressurized and refrigerated subjects are considered. No account is taken of process response to the release. If the user prefers, the release rate can be specified as input data. The releases might result in a BLEVE (Boiling Liquid Expanding Vapour Explosion), a poolfire, a jetfire, a sprayfire, a flashfire, an explosion inside a building, a vapour cloud explosion or spreading of toxic gases or any combination of these.

The estimation of the consequences of an accidental release is based on simple well known models. The evaluation of a scenario can be characterized as a dialog between the user and the computer. The system asks some questions and then runs a small calculation program, depending on the result new questions are asked and so on. A short description of the evaluation process is given below.

First the user has to choose an initial release. You can choose an instantaneous and continuous or a "user defined" continuous release. Since no sharp distinction exists between instantaneous and continuous releases a check of this selection is made at the end of the evaluation. The appropriate questions are asked so that concentration and physical properties in the nearfield can be estimated. Then what kind of hazard this release results in has to be estimated. It can result in a poolfire, a jetfire, a sprayfire, a flashfire, an explosion, a toxic hazard etc. After this estimation the effect on human beings and components has to be evaluated. The effect of toxic gases, radiation, pressure and flying missiles are considered. If the initial release results in a fire then a jump to the fire part of the system is done. The result is presented as estimated time to damage of components and human hazards.

The main difference between this expert system and other consequence analysis programs is that when using other programs you have to be very familiar with the modelling of accidental releases because no warnings are given when the outside of the validation limits of the models is used. In this expert system warnings are given and the system gives an estimation of the properties that are currently calculated. Also this program takes releases inside a building into consideration, but it does not evaluate the concentration field inside a building due to a release outside the building. Estimating damage on a component is also a feature that other programs do not contain.

Fire in a compartment

The calculation procedures contained in the system for fires in compartments are divided in two categories:

Part 1  Computes the environment parameters. Here, consideration is given to ventilation openings, heat conducted to walls, oxygen concentration and output development, heat and smoke detector’s resistance, mechanical ventilation and numerous other physical processes.

Part 2  Once the fire environment is known, the effect of this environment on components in the compartment can be estimated. A first step is to calculate
the incident heat flux from the flame and the hot gas layer, and the convective heat transfer from the gas layer or the plume, to the component. A second step is to compare the external heating experienced by the component to the known heating criteria which will lead to damage of the component.

Knowing the temperature and the height of the gas layer allows the view factor between it and the component, and thus the incident heat flux, to be calculated. Similarly, assuming the flame to have the shape of a cylinder (its height and diameter given by Fire Simulator), the view factor between the flame and the component can be calculated and hence, the incident heat flux.

The second step, i.e. comparing external heating to damage criteria, is more difficult. For components of simple shape and construction, solutions of the general heat conduction equation can approximate how component surface temperature changes with time. This can be compared with experimentally or otherwise determined critical surface temperature values and a time to component damage can be estimated.

For more complex components, direct comparisons with experiments are needed and for some components such experiments are scarce. This can be mentioned as "expert advice", where conservative estimates are made with regard to time to component damage, the accuracy of this estimate being closely linked to the quality and amount of experimental data available.

The packaged calculation model Fire Simulator has been used. Fire Simulator is part of the PFETOOL program package developed by the National Institute of Standards and Technology (NIST) in the U.S.A.

Fire Simulator solves two simultaneous differential equations, based on the concept of conservation of energy and conservation of mass, to compute the temperature and volume of the smoke layer produced by the prescribed rate of heat release. One equation calculates the increase in temperature considering the energy input, the energy losses and the air entrained by the rising plume. The other calculates the rate of decent of smoke layer as a function of the increase in temperature and the rate of mass flow in the rising plume at the point where it meets the descending smoke layer.

Additionally, certain modules from FPETOOL are used to draw graphs of the results, to help the user define an energy release rate, to allow the user to build and use databases, etc. The concentration of unburnt hydrocarbons leaving the enclosure is also calculated, allowing the risk of fire spread to an adjacent compartment to be estimated.

The above methodology can be used to estimate time to flashover in a compartment and the risk of a second compartment becoming involved, time to detector or sprinkler activation, the time to damage of critical components such as pumps and electrical
cables, the influence of mechanical ventilation and smoke relief vents, etc, etc. Such information can then be used by engineers to help make decisions on fire protection measures to be implemented in power plants and other industrial installations.

**Natural gas accident in Falkenberg, Sweden**
A natural gas accident occurred in Falkenberg, Sweden in early October 1993. An excavating machine tore a 3-4 centimeter hole in a natural-gas line. The machine operator reacted quickly and turned off the machine and then hurriedly departed the site accompanied by the other members of the work team. They stopped traffic on nearby roads and alerted the rescue service. The gas never ignited and the accident had a fortunate outcome.

But what would happened if the gas had ignited?

FREIA was fed input data and a computation was made of the consequences of an ignition.

Since natural gas is light, it ascends rapidly and combines with the air. If ignition is immediate and a jet flame is formed, the effect is considerably greater. According to the computations, the jet flame would have been nine meters high.

The radiation from the jet flame would cause different burn injuries at people at the distance shown in table 1.

<table>
<thead>
<tr>
<th>Burn injuries</th>
<th>Distance in jet flame direction (m)</th>
<th>Distance bidirectional to the jet flame (m)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Third degree (burns requiring professional care and skin transplants)</td>
<td>11</td>
<td>4</td>
</tr>
<tr>
<td>Second degree (blisterforming, redness and pain, 2-4 week healing period)</td>
<td>12</td>
<td>5</td>
</tr>
<tr>
<td>First degree (stinging with skin redness)</td>
<td>14</td>
<td>8</td>
</tr>
</tbody>
</table>
OECD Nuclear Energy Agency

Meeting on Fire Protection and Fire Protection Systems in Nuclear Power Plants

Cologne, Germany, 6th to 9th December 1993

Model of filter clogging in case of fire

J.C. LABORDE, C. PREVOST, J. VENDEL

Institut de Protection et de Sûreté Nucléaire
Département de Protection de l’Environnement et des Installations
Service d’Études et de Recherches en Aérocontamination et en Confinement
IPSN/CEA - 91191 Gif-sur-Yvette Cedex

Abstract

One of the main roles of ventilation in a nuclear plant is to maintain dynamic confinement during normal or accidental operating conditions. Among the incidents likely to affect a nuclear facility, fire is one of those which, coming from the safety standpoint, requires the greatest attention because it is one of the most probable risks.

Studying fires and their consequences is complicated due to the diversity of the causes and parameters easily be conceived that thermal challenges will, in almost cases, be associated with other challenges such as the release of gases and aerosols, pressure stresses and vapour condensation.

To evaluate the consequences of a fire and to develop strategies, the Airborne Contamination and Confinement Research Group (IPSN/DPEI/SERAC) initiated a research and development program including experimental and modeling studies.

The paper presents an empirical model describing the clogging of High Efficiency Particulate Air (HEPA) filters by the aerosols derived from the combustion of standard materials used in the nuclear industry.

It gives the results obtained from laboratory tests, relative to the determination of the variation of the air flow resistance of a filter clogged, according to the masses of deposited aerosols. This empirical model can then be used in calculation codes describing the consequences of fires in the confinement of a nuclear facility.
I Introduction

Of the various phenomena associated with fires in nuclear facility safety evaluations, the presence of aerosols is one of those that requires the greatest attention, because it can alter the ventilation and air cleaning functions.

That is, fire is not just a thermal phenomenon. The main parameters associated with it may be summarized as follows:

![Diagram of fire components]

The presence of aerosols is therefore a parameter to be taken into account. It has the effect of clogging the High Efficiency Particulate Air (HEPA) filters used in the ventilation networks. This clogging may have consequences affecting the containment of radioactive substances within the nuclear facility, by:

- decreasing the exhaust flow rate from any area equipped with clogged HEPA filter. This can lead to an overpressure in the area affected by the fire.
- breakage of the sheets of filter material constituting the HEPA filter medium, considering their mechanical strength characteristics and the high temperatures involved. The filter then loses much of its efficiency.

In order to predict HEPA filter clogging and develop a model that can be used in a system of containment simulation codes, experiments are being conducted on different test rigs at the Airborne Contamination and Confinement Research Group, in which filter clogging is evaluated as a function of the aerosols formed by the burning of materials typically used in the nuclear industry.
II  Experimental facilities

Two test facilities are currently available.

Beatrice test facility

This is a full scale facility for testing filters of standard dimensions and design (those with pleated filtering medium). The test rig consists of a stainless steel enclosure measuring 2 m$^2$, in which the combustible materials are placed, along with an exhaust conduit of rectangular cross section, in 2 m lengths.

The exhaust circuit includes two fans in parallel, one or the other of which can be used to obtain the desired exhaust flow rate, which is 3,400 m$^3$/h for a full filtering cell, or 600 m$^3$/h for a single dihedra. The filters to be studied for clogging are installed upstream of the fans. This facility is equipped with the instrumentation needed to acquire the following data: temperature at the HEPA filter, size distribution and concentration of aerosols formed, filtration flow rate and filter pressure drop.

Downscale test facility

This is a laboratory system for working with plane filters, in order to extend the range of validity of the clogging curves generated on the Beatrice rig. The upstream portion of this rig consists of an enclosure of 30 dm$^2$, connected to an exhaust pipe where the air is exhausted through ejector nozzles; the rated flow rate of this facility is 10 m$^3$/h.

III  Combustible material used and HEPA filters tested

Combustible Materials

The fire tests were performed on four types of polymer material:

- PMMA  methyl acrylathe polymer
- PVC  polyvinyl chloride, of three types:
  . pink vinyl
  . opaque
  . transpare
- PS  polystyrene

These materials are used either in the pure state or mixed.

The combustibles are samples measuring about 100 cm$^2$ and placed in a pan containing alcohol to start them burning. The type and weight of these combustibles is varied from one test to the next. Most of the fire tests were performed with oxygen in excess of stoichiometric proportions, but some were performed under oxygen depletion conditions.
HEPA filters

The HEPA filters used are of two types:

- plane filters consisting of a glass fiber medium, with a filtration area of about 100 cm²;
- mini-pleats filters with a filtration area of 36 m² for a full filter cell and 7 m² for a single dihedra.

IV Results

The purpose of the tests was to determine the variation in the ventilation airflow resistance $R$ of the filter as a function of the mass $M_d$ of aerosols deposited on the filter.

$R$ is defined by the relation

$$ R = \frac{\mu_0 \Delta P}{\mu Q_v} $$

$\Delta P$ : filter pressure drop
$\mu$, $\mu_0$ : dynamic viscosity of gas under experimental and atmospheric temperatures, respectively
$Q_v$ : volumetric flow rate through the filter

A typical example of the results obtained with PMMA fires is given in figure 1, where the variations of $R/R_0 = f(M_d)$ are shown for different types of tests.

$R_0$ is the filter resistance of the unclogged filter.

![Graph](image)

Figure 1 - Variation of $R/R_0$ as a function of $M_d$ for PMMA fires.

Note
FC : test with a full filter cell
PF : test with a plane filter
The clogging curves vary according to the type and composition of the material burned. As an example, figure 2 gives the variations $R/R_0 = f(M_d)$ for a mixture of PMMA and transparent PVC, in different proportions.

Figure 2 - Effect of the composition of the burned mixture on the variations of $R/R_0$.

In the pure state, it is PMMA that has the greatest clogging power, and for a given mixture, the most penalizing proportions seem to be those at which PMMA is largest in quantity. In those tests where less PMMA is present, and in those where the pure material is difficult to set to flame, the thermal degradation of the material is a pyrolysis rather than open combustion because of the insufficient thermal flux added. The aerosol formation kinetics must therefore be different, which results in less clogging power, due to the nature of the aerosols formed.

V Empirical clogging model

Generally speaking, for a given material, we observe that the tests on this material can be correlated with each other. The correlation function $f$ must satisfy the initial condition:

$$
\text{for } M_d = 0 \quad \frac{R}{R_0} = f(0) = 1
$$
A type of function common to all of the materials, satisfying the initial condition, and integrating the variation as best possible, is:

\[
\frac{R}{R_0} = 1 + a (M_d)^b
\]  

(1)

in which \(a\) and \(b\) are constants characteristic of the material and of the thermal degradation considered.

Relation (1) is purely empirical, \(a\) and \(b\) being determined from smoothing the curve of experimental points with a power function. The complexity of the type of aerosols formed (solids + liquids, because of condensation effects) in the course of a fire and the knownledges about HEPA filter clogging, do not allow us to determine the resistance of a clogged filter from purely theoretical considerations. Modellings of the solid aerosols have nonetheless been developed by various authors [1], [2].

Table 1 gives the constants \(a\) and \(b\) as well as the variation interval \([0, M_d]\), in which the proposed relation was validated, for several materials tested.

<table>
<thead>
<tr>
<th>Material</th>
<th>Thermal degradation</th>
<th>(a)</th>
<th>(b)</th>
<th>variation interval (g/m²)</th>
</tr>
</thead>
<tbody>
<tr>
<td>PMMA</td>
<td>(O_2) Excess</td>
<td>0,71</td>
<td>1,20</td>
<td>0 - 16</td>
</tr>
<tr>
<td></td>
<td>(O_2) Depletion</td>
<td>0,61</td>
<td>1,04</td>
<td>0 - 16</td>
</tr>
<tr>
<td>Polystyrene</td>
<td>(O_2) Excess</td>
<td>0,73</td>
<td>0,86</td>
<td>0 - 65</td>
</tr>
<tr>
<td></td>
<td>(O_2) Depletion</td>
<td>0,08</td>
<td>1,10</td>
<td>0 - 60</td>
</tr>
<tr>
<td>Pink PVC</td>
<td>(O_2) Excess</td>
<td>0,11</td>
<td>0,26</td>
<td>0 - 10</td>
</tr>
<tr>
<td>Opaque PVC</td>
<td>(O_2) Excess</td>
<td>0,06</td>
<td>0,73</td>
<td>0 - 4</td>
</tr>
<tr>
<td>9% PMMA + 1% transparent PVC</td>
<td>(O_2) Excess</td>
<td>0,5</td>
<td>1,26</td>
<td>0 - 25</td>
</tr>
<tr>
<td>9% PMMA + 1% transparent PVC</td>
<td>(O_2) Excess</td>
<td>0,5</td>
<td>1,26</td>
<td>0 - 25</td>
</tr>
<tr>
<td>1% PMMA + 9% transparent PVC</td>
<td>(O_2) Excess</td>
<td>0,006</td>
<td>1,6</td>
<td>0 - 50</td>
</tr>
<tr>
<td>9% PMMA + 1% opaque PVC</td>
<td>(O_2) Excess</td>
<td>0,32</td>
<td>1,22</td>
<td>0 - 30</td>
</tr>
<tr>
<td>1% PMMA + 9% opaque PVC</td>
<td>(O_2) Excess</td>
<td>0,003</td>
<td>2</td>
<td>0 - 85</td>
</tr>
<tr>
<td>9% PMMA + 1% opaque PVC</td>
<td>(O_2) Excess</td>
<td>0,018</td>
<td>1,65</td>
<td>0 - 60</td>
</tr>
<tr>
<td>9/10 PMMA + 1/10 Pink PVC</td>
<td>(O_2) Excess</td>
<td>0,69</td>
<td>1,05</td>
<td>0 - 8</td>
</tr>
<tr>
<td>4/5 PMMA + 1/5 Pink PVC</td>
<td>(O_2) Excess</td>
<td>0,5</td>
<td>0,95</td>
<td>0 - 5</td>
</tr>
</tbody>
</table>

Table 1

Note

We were not able to obtain identical \(M_d\) variation intervals for all of the tests, because some of them caused a degradation of the HEPA filter (mechanical breakage of the filtering medium element sheets for standard size filter cells) and had to be interrupted before all of the material was burned.
VI Characterization of the aerosol source term

In practical applications of the clogging model, it is necessary to know the mass of aerosols $M_a(t)$ deposited on the filter at any given time.

If $Q_m(t)$ is the mass flow rate of aerosols formed during the burning of a material, and $Q_f(t)$ the fire heat release rate of the combustible as a function of time, we have:

$$M_a(t) = \int_0^t Q_m(t) \, dt$$

The fire source term therefore includes a thermal part that corresponds to the formation of the gases by combustion or pyrolysis, generating a fire heat release rate $Q_f(t)$, and an aerosol part that corresponds to the formation of aerosols in the course of the fire. For the clogging model, it is this aerosol source term that should be taken directly into account. There are two possible ways of entering this term as input data to the model:

- directly from $Q_m(t)$ deduced from experiments.

This curve is:

$\begin{array}{c}
\text{\bf Q}_m (t) \\
\text{t}
\end{array}$

The curve maximum corresponds to maximum mass loss rate of the combustible.

- indirectly from the curve $Q_f(t)$.

To do this, we assume that the mass flow rate of aerosols can be related to the fire heat release rate, in particular when there is an excess of air with respect to stoichiometric proportions.

Then, letting:

$\Delta m(t)$ be the mass loss of the combustible,
$m_g(t)$ the mass of combustion gases formed,
$m_a(t)$ the mass of aerosols formed,
$k$ the quantity of aerosols formed with respect to the mass of material burned (generally $k < 0.1$),
we can say that:

\[ \Delta m (t) = m_k (t) + m_a (t) \]

\[ = \Delta m (t) = m_k (t) \]

and \( \frac{d \Delta m (t)}{dt} = \frac{d m_k (t)}{dt} \)

whereas

\[ Q_T (t) = h_c \frac{d \Delta m (t)}{dt} \text{ with excess air,} \]

\( h_c \) : combustion heat

whence

\[ Q_m (t) = \frac{k Q_T (t)}{h_c} \quad (2) \]

Experimentally, when the \( Q_m(t) \) and \( Q_T(t) \) curves are compared, we find that they have the same shape.

VI.1. Mass flow rate of aerosols

As an example, the results obtained for a given material (case of PMMA) are given in figure 3 in the form:

\[ \frac{Q_m}{M_i} = f (t) \]

in which \( Q_m \) is the mass flow rate of aerosols formed and \( M_i \) the mass of material actually burned.

Figure 3 - Variation of \( Q_m/M_i \) as a function of time
It is observed that, considering the dispersion of the experimental data, it is impossible to correlate the results for various experiments carried out on the same material, including when the tests are performed on the same test facility under similar thermal degradation conditions. This very surely is due to the difficulty there is in reproducing identical operational conditions on full scale test facility like Beatrice. That is, parameters like the material geometry (sample shape and size) and the arrangement of the samples in the test pan must have an effect that is difficult to control. On Beatrice, the samples of the various materials are arranged randomly from one test to the next.

This difficulty in finding an acceptable correlation among the various aerosol mass flow rates therefore led us to check if there existed any correlation between the aerosol mass flow rate and the fire heat release rate for a given test.

VI.2 Correlation between the aerosol mass flow rate and the fire heat release rate

It can be verified from a few tests, for which $Q_m(t)$ and $Q_r(t)$ are available, whether or not the approach explained above is realistic. These are the fire tests of:

- PMMA
- (2/3)PMMA + (1/3) transparent PVC
- (9/10)PMMA + (1/10) pink PVC.

The curves representing the instantaneous variations of the ratio $r = Q_r(t)/Q_m(t)$ for these tests are given in figure 4.

![Figure 4 - Instantaneous variations of r](image-url)
These results show that, under steady thermal degradation conditions, the value \( r \) varies little from its average value, for a given material. We get the following results:

<table>
<thead>
<tr>
<th>Materials</th>
<th>( r ) (kJ/kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>PMMA</td>
<td>( 3.6 \cdot 10^6 \pm 1.4 \cdot 10^6 )</td>
</tr>
<tr>
<td>9/10 PMMA + 1/10 pink PVC</td>
<td>( 7.4 \cdot 10^5 \pm 2.4 \cdot 10^5 )</td>
</tr>
<tr>
<td>2/3 PMMA + 1/3 transparent PCV</td>
<td>( 1.08 \cdot 10^5 \pm 1.6 \cdot 10^4 )</td>
</tr>
</tbody>
</table>

These preliminary results hint at the possibility of correlating the aerosol mass flow rate with the fire heat release rate. Further tests should be carried out to define the domains of validity of this approach and to determine the ratio \( r \) for other materials.

VII Conclusion

In evaluating the safety of nuclear facilities, the presence of aerosols is one of the fire related phenomena requiring the greatest attention because it can generate alterations in the ventilation and air cleaning functions and therefore in the containment of radioactive substances. For this reason, it must be taken into account in the development of a system of fire simulation codes.

This is why a clogging model has been developed. To use this model, the variation of the resistance of a clogged filter (\( R \)) as a function of the mass of deposited aerosol, \( (M_d) \) must be known. Considering the complexity and diversity of fires that can occur and the diversity of aerosols thereby formed, an empirical approximation was made from various experiments data.

The result is that the ventilation resistance \( R \) can be found from the initial resistance \( R_0 \) of an unclogged filter and from the mass of aerosols deposited by the relation:

\[
\frac{R}{R_0} = 1 + a (M_d)^b
\]

in which \( a \) and \( b \) are constants characteristic of the material and of the type of thermal degradation considered.

Moreover, it seems possible to find the mass of aerosols deposited on a filter from the variations of the fire heat release rate. That is, the aerosol mass flow rate may be directly related to the fire heat release rate under steady thermal degradation conditions.

Lastly, the effects of certain parameters (filtration velocity, combustion temperature, type of thermal degradation) can be investigated in laboratory test facilities to improve our knowledge of the clogging mechanisms and refine the proposed empirical approximations.
REFERENCES

[1] P. LETOURNEAU, V. RENAUDIN, J. VENDEL
Effects of the particle penetration inside the filter medium on the HEPA filter pressure drop
- 22nd Nuclear Air Cleaning Conference, Denver 1992

Enhanced filtration program at LLNF. A progress report - 15th Nuclear Air Cleaning
Conference, Boston 1978
OECD Nuclear Energy Agency

Meeting on Fire Protection and Fire Protection Systems in Nuclear Power Plants

Cologne, Germany, 6th to 9th December 1993

Calculation code evaluating the confinement of a nuclear facility in case of fires

J.C. LABORDE, C. PREVOST, J. VENDEL

Institut de Protection et de Sûreté Nucléaire
Département de Protection de l'Environnement et des Installations
Service d'Études et de Recherches en Aérocontamination et en Confinement
IPSN/CEA - 91191 Gif-sur-Yvette Cedex

Abstract

Accident events involving fire are quite frequent and could have a severe effect on the safety of nuclear facilities. As confinement must be maintained, the ventilation and filtration systems have to be designed to limit radioactive release to the environment. To determine and analyse the consequences of a fire on the contamination confinement, a research program including the development of calculation code and experimental studies has been carried out, in France, at the Airborne Contamination and Confinement Research Group (IPSN/DPEI/SERAC).

The paper describes a calculation code developed in order to evaluate the behaviour of confinement barriers encountered in the ventilation network of a nuclear facility. Its permits also to know the flowrates, temperatures and pressures at the different points of the network.

This calculation code results from the coupling of a ventilation code (SIMEVENT) with several models describing the temperature in a room resulting of a fire, the temperatures along the ventilation ducts, the contamination transferes through out the ventilation equipments (ducts, dampers, valves, air cleaning systems) and the HEPA filters clogging.

Then, the paper gives an application on a simple ventilation network.
I. Introduction

Within the scope of nuclear protection and safety, integration of the fire risk in a nuclear facility mainly involves:

- evaluating the consequences of fire on the confinement and determining the source term released to the environment,

- defining strategies, basically through ventilation control, aimed at reducing these consequences.

In view of the complexity of the problem raised, it is indispensable to use calculation codes integrating the descriptions of the ventilation network and fire, associated to experiment programs.

In collaboration with various teams from the CEA group, the Airborne Contamination and Confinement Research Group has participated in developing a calculation code using the SIMEVENT ventilation software which comprises various modules associated to fire risk and mass transfer in the ventilation networks.

This paper presents the current level of progress made on this code and an application example on a simple installation.

II. SIMEVENT calculation code

The SIMEVENT calculation code has already been presented in various international conferences [1], [2].

It is used to evaluate the behaviour of a complex ventilation network submitted to various disturbances of mechanical and/or thermal origin. The code is based on the division of the system in nodes linked by branches. A node represents a point of the system whose temperature and pressure can be considered as uniform. A branch is a part of the system limited by two nodes and fully defined by the relation \( \Delta P = f(Q) \) which expresses the pressure difference (\( \Delta P \)) as a function of the flow rate (\( Q \)) in the branch. A branch may be any kind of classical ventilation element (blower, valve, duct, filter, ...). The fluid mechanics constitute laws of these elements are memorized in the SIMEVENT library.

For a given situation, the code makes it possible to calculate the new values of pressure and temperature at the nodes and the flow rates in the branches. It can then be seen whether the simulated incident causes undesirable effects with respect to safety (reversal of air flow, room overpressure, etc).

An extension to this code has been proposed to integrate the risk of fire and mass transfer (gases and aerosols) in the ventilation networks.
III. Extension of SIMEVENT code to fire

The extension of the SIMEVENT code to fire implies coupling with a code capable of describing the development of a fire in a room and introduction of models describing the effects associated to the fire (clogging of HEPA filters, thermal loss in ducts, mass transfer).

In particular, the presence of aerosols is a phenomenon which cannot be dissociated from the purely thermal effect of a fire, which specifically results in clogging the HEPA filters installed in the ventilation networks. This phenomenon must therefore be included in the SIMEVENT + FIRE code.

The thermal and mechanical effects in the SIMEVENT code can be schematized as follows:

\[ Q_f(t) \quad \text{heat release rate generated by combustible (variable in time),} \]

\[ Q_m(t) \quad \text{mass flow of aerosols formed,} \]

\[ M_d(t) \quad \text{weight of aerosols deposited on filter,} \]
\[ M_d(t) = \int_0^t Q_m(t) dt \]

R(t): aeralic resistance of clogged filter

\[ Q_f(t) \quad \text{and} \quad Q_m(t) \quad \text{are input data of the codes used.} \]

III.1. Fire code

The approach presented here is based on a coupling of the SIMEVENT code with a fire code based on a simplified model giving the temperature rise for forced ventilated compartment fires (one-zone model). This temperature correlation has been developed at the Lawrence Livermore National Laboratory (LLNL) [3].

A more accurate analysis can be obtained if necessary by coupling SIMEVENT with a fire code based on a multi-zone model, such as the FLAMME code [4].

The empirical correlation elaborated at the LLNL estimates the upper layer temperature rise above ambient by:

\[ \frac{T - T_a}{T_a} = 0.63 \left( \frac{Q_f}{m_a C_{pa} T_a} \right)^{0.72} \left( \frac{h A}{m C_{pa}} \right)^{-0.36} \]

where:
- T: upper layer temperature (K)
- T_a: ambient temperature (K)
- Q_f: fire heat release rate (kW)
- C_{pa}: gas specific heat capacity (kJ.kg^{-1}.K^{-1})
- m_a: mass ventilation flow rate (kg.s^{-1})
- A: compartment surface area (m^2)
- h: effective heat transfer coefficient (kW.m^{-2}.K^{-1})
\[ h = \sqrt{\frac{k_m \rho_m C_{pm}}{t}} \quad \text{if} \ t < t_p \]

\[ h = \frac{k_m}{e} \quad \text{if} \ t_p \geq t \]

where:

- \( k_m \): wall specific heat capacity (kJ kg\(^{-1}\) K\(^{-1}\))
- \( \rho_m \): wall density (kg m\(^{-3}\))
- \( e \): wall thickness (m)
- \( t_p \): wall thermal penetration time (s)

\[ t_p = \left( \frac{\rho_m C_{pm}}{k_m} \right) \left( \frac{e}{2} \right)^2 \]

\( t \): time (s).

This equation has been found to be extremely accurate in predicting upper layer temperatures in the test cell of LLNL with standard fire source of methane gas (approximately constant heat release).

Other comparisons have been made from data collected on tests cells in which polymer fires are performed [5]. These fires are characterized by heat releases \( Q_t \) variable with time.

### III.2. Clogging model

In SIMEVENT, the filter clogging is integrated using the following model:

\[
\text{FILTRE (A, B, C) = } R_{o}, m_{d1}, R_{1}, m_{d2}, R_{2}, \ldots m_{dn}, R_{n}
\]

where:

- \( A \) and \( B \) are the nodes between which the filter is installed.
- \( C \) is the node of the fire room (by default, \( C = A \)).
- \( R_{o} \) is the unclogged filter resistance.
- \( R_{i} \) is the filter resistance for a weight \( m_{d_{i}} \) of deposited aerosols.
This FILTRE model generates the following relation:

$$\Delta P_i = R_i \cdot \frac{\mu}{\mu_o} \cdot Q_v$$

$\Delta P_i$: filter pressure drop,
$\mu, \mu_o$: dynamic viscosity of gas in filter at T and ambient temperatures, respectively,
$Q_v$: volumetric flow rate at the filter.

The quantity $m_{d_i}$ of aerosols deposited in time is calculated using the fire source term as indicated above.

Owing to the complexity of the aerosols formed (solid particles + liquids due to condensation effects) during a fire and the current knowledge relative to HEPA filters clogging, it is not possible at this time to determine, through a purely theoretical approach, the value of the aeraulic resistance of a clogged filter. This is why the resistance $R_i$ is introduced in SIMEVENT using an empirical approach, also presented in this conference [6].

III.3. Thermal losses

The SIMEVENT code applied to fire has models used to evaluate the thermal losses occurring through the walls of the rooms and along the duct walls using classic heat transfer laws. Some of these models translate the heat transfers according to basic mechanisms (conduction, convection, radiation).

A special model, called *TUAYAU, covers all the mechanisms relative to thermal losses occurring along ducts.

* TUYAU ($N_o, N_1, N_2) = L, D_i, \Delta, S, e, \lambda, h_e$

$L$: length of duct
$D_i$: inside hydraulic diameter
$\Delta$: relative roughness
$S$: cross section
$e$: thickness
$\lambda$: thermal conductivity of wall
$h_e$: external exchange coefficient
$N_1, N_2$: upstream and downstream nodes delimiting ducts
$N_0$: ambient node with which duct exchanges heat
This model is used to calculate the thermal losses along the duct between $N_1$ and $N_2$ and therefore the temperature at the duct outlet (node $N_3$).

IV. Mass transfer in ventilation networks

To calculate the concentrations of contaminants transferred to various points of the installation and released to the environment, the SIMEVENT code has been modified, by introducing the required models, to take into account the mass transfers in both gas and particle form.

The modifications proposed comprise:

- weight balances at the various nodes of the network indicating the weight conservation of the concerned type,
- weight balances in the branches of the network indicating the mass deposits in the element forming the branch itself.

\[ C_e \quad \frac{\text{branch}}{\text{C}_s} \]

As a general rule: \( C_s = k \ C_e \)

$C_e$: concentration at the branch outlet  
$C_s$: concentration at the branch inlet  
$k$: deposit function

\[ k = f (\text{flow}, \quad \text{branch,} \quad \text{material}) \]

\[ \downarrow \quad \downarrow \quad \downarrow \]

- flow rate  
- duct  
- gas  
- physical properties  
- filter  
- aerosol  
- register  
- room

The deposit function can be entered either by the user or calculated by a specific model of the element forming the branch.

At this time, two specific models are available:

- the deposit function when the branch is a duct (KCONDUIT),  
- the deposit function when the branch is a filter (KFILTRE).

These two models are based on the basic mechanisms relative to the deposit of an aerosol on a wall and on the basic mechanisms relative to filtration.
For example purposes, the input data and results provided by the KCONDUIT function are summarized below:

**INLET**

- Aerosol
  - size distribution:
    - median diameter
    - geometric standard deviation
  - total numeric or mass concentration
  - density

- Flow
  - flow rate
  - temperature, pressure
  - physical properties of fluid

- Duct
  - shape (horizontal, vertical, slanted, elbows)
  - dimensions

**OUTLET**

- total numeric or mass concentration
- concentration per range of diameter

**KCONDUIT**

**Note:**

At this time, the mass transfer in SIMEVENT does not take into account the possible changes in phase of the concerned material.

V. General structure of SIMEVENT code applied to fire

The general structure of the SIMEVENT code and of the various models used as part of the studies related to fire is presented below:
VI. Example of application to a simple facility

The facility used for the example (MELANIE test facility [5]) comprises a single room with its associated ventilation network (supply and extraction network with HEPA filter). The SIMEVENT model of the facility is shown in figure 1.

Figure 1: SIMEVENT model of the facility
The SIMEVENT model involves a fire using a mixture of PMMA (methyl acrylate polymer) + PVC (polyvinyl chloride). The values calculated from the SIMEVENT code applied to the fire then compared to the experimental values are as follows:

- extraction flow rate (Q),
- pressures in room and upstream of HEPA extraction filter (P),
- temperature in room (T),
- concentration of aerosols upstream of HEPA extraction filter (Ca).

The results obtained are given below. Note the good correlation between the results given by the code and the experiment. The results underscore, in particular, a risk of overpressurization of the room, which is a phenomenon to be avoided in a nuclear safety program.

VII. Conclusion

The evaluation of radioactive contamination confinement capacity requires knowledge of the parameters (in particular, temperature, pressure, pollutant concentration) defining the overall state of the ventilation network when a fire breaks out at a given point in the network.

The improvements made to the initial version of the SIMEVENT calculation code provide the required information. In this respect, the SIMEVENT code has been modified by introducing models integrating the various phenomena associated to fire risks: increased temperature in room, thermal losses in ducts, clogging of HEPA filters, transfers of particles and gases in various parts of the network.

With this new version of the code, it is thus possible to verify the risk of a room becoming overpressurized, determine the thermal and mechanical stresses acting on the HEPA filters, calculate the quantities of radioactive substances released. All of these elements are of fundamental importance in evaluating the safety of a nuclear facility.

Furthermore, it is possible to study the influence of the ventilation control (opening or closing of fire dampers) on the system capacity to maintain confinement.

Development of the code is being continued, at this time, to include aerosol deposits in other elements of the network (balancing registers, fire dampers) and to introduce changes in phase of certain composites (gas condensation, for example).
REFERENCES

[1] G. MARTIN, Ph. MULCEY, Ph. PERDRIAU, Ph. PRUCHON, S. RABOIN
Simulation of ventilation networks: presentation of the safety code PIAF complementary
with fluid mechanics codes. 19th Nuclear Air Cleaning Conference, Seattle 1986.

[2] J.C. LABORDE, Ph. MULCEY, Ph. PERDRIAU, S. RABOIN
Simulation of ventilation networks using the SIMEVENT CAD System. Ventilation' 91

Temparature correlation for forced ventilated compartment fires. 1st International Symposium
on fire safety science. Gaithersburg 1985.

Calculating the consequences of a kerosene pool fire: the FLAMME computer code.
Interaction of fire and explosion with ventilation systems in nuclear facilities, Los Alamos
1983

Caractérisation et conséquences de feux de polymères en local ventilé. Evaluation des

Model of filter clogging in case of fire. Meeting on fire protection and fire protection
systems in nuclear power plants - Cologne, december 1993.
Some Results of CEC-Standard Problem Predictions of Effects caused by a Cable Fire Experiment within the HDR-Facility

H. Karwat
Technische Universität München
Lehrstuhl für Reaktordynamik und Reaktorsicherheit

Communication to the OECD Specialist Meeting on Fire Protection and Fire Protection Systems in Nuclear Power Plants
Cologne, Germany, December 6.-9., 1993
1 THE OBJECTIVES OF THE STANDARD PROBLEM EXERCISE

A highly effective way to increase the confidence in the validity and accuracy of code predictions is achieved by comparing a variety of experimentally measured parameters with the calculated predictions of several analytical simulation models normally used to predict such events. Since many years such comparative exercises are performed under the umbrella of international organisation, e.g. the Commission of the European Community (CEC) or the Organisation for Economic Co-Operation and Development (OECD). Both organisations have a long tradition in organizing and executing such activities, termed International Standard Problems. The benefit of this activity is seen in the fact that the results of expensive experiments are offered to an international community of code users which give them a chance to calibrate their way of using the codes against the reality of an experimental background /OEC89/.

With the agreement of the German Federal Ministry for Research and Technology (BMFT) the HDR-Project has offered the cable fire experiment E42.2 as a basis for an International Standard Problem to be performed under the sponsorship of the Commission of the European Communities (CEC). The experiment has been selected from the point of view of its interest for the nuclear community as well as from the interest existing within the non-nuclear fire protection community.

The HDR test facility has been described in detail on occasion of several international standard problems (ISP-16, ISP-23 and ISP-29) /SCH82, SCH82a/. A vertical cross section of the test facility is shown in fig. 1. A designated section of the containment at the 4,5 m elevation has been specifically prepared to perform combustion experiments (fire compartment) /MUL92/. The fire compartment is shown by fig. 2 which depicts a horizontal cross section of the HDR containment. The specific fire compartment has a total volume of 102,5 m³ and a floor area of 24,1 m². Details of the fire compartment are shown in fig. 3. The fire compartment consisted of 2 rooms (I and II) with a volume of 102,5 m³ separated from two other rooms (III and IV) with a total volume of 109,2 m³ by a two-wing door of (2x0,92x1,98) m² flow area. One wing was locked closed while the second wing could be opened during the experiment. Room No. III was connected by a (0,5x2,45) m² orifice near the floor with the general containment natural ventilation system to allow for natural convection driven fresh air supply after opening one wing of the door between rooms II and III.

The specific properties of the confining structures of the fire compartment have been communicated by the task specification for the analytical prediction of the effects caused by the
cable fire experiment E42.2 /KAR92/ . Specific provisions have been made to avoid a far reaching contamination of the HDR containment by aerosols possibly containing dioxine. Hence, a specific ventilation system has been installed equipped with highly efficient filter units. The flow sheet of the air ventilation systems of the fire compartment is shown in fig. 4. The air exchange rate of the fire compartment may be varied between 5 and 17 air changes per hour. Fresh air supply into the fire compartment was possible through a blower unit VB191 as well as through free convection from the rest of the containment through a wall opening of approx. 1,25 m² as mentioned above. Exhaust of air and of combustion gases was possible mainly by 3 specifically installed exhaust lines connected to filters of high efficiency and blowers. Only the exhaust blower VB190 was in operation during the experiment.

Within the fire compartment 3 cables racks have been installed which were loaded with various types of selected cables. Another 3 racks were installed vertically fixed to a side wall of the fire compartment. Fig. 5 shows the local position of the 6 cable racks as well as the major positions of temperature sensors strings A to J of the fire compartment and of the room in front of the fire compartment. More details concerning the location of individual thermocouples and the structure of the different cables and the material properties have been provided within the task specification /KAR92/. Fig. 6 shows the vertical arrangement of the cable trays, the distribution of cables types and the position of the heat protection layers of ALSIFLEX on the installed cable racks.

A gas burner was operated for 60 s igniting the adjacent cable package located adjacent to the cable rack I as indicated on fig. 6.

Cable racks I and II were mounted on 4 load cells each to measure the weight loss of both racks during the experiment.

2 MAIN FEATURES OF THE EXPERIMENT AND THE ANALYTICAL SIMULATION TASK

The cable fire has been executed on January 22, 1992, it was initiated by starting the gas burner below channel 5 at rack I. According to the specification channels 1-5, 12-13 and 23 were supposed to be involved in this experiment only. The cables on rack III (channels 20-22) have been burnt during the preceding experiment E42.1 and were not available for combustion during this experiment. The air supply conditions for the fire compartment were
changed upon the indication that at least 50% of the cables installed on channels 1-5 of rack I had been ignited. Actually this happened around 8 min after starting the experiment. One wing of the two-wing door to the fire compartment was opened remotely at 8 min after starting the experiment. This caused a considerable enhancement of the combustion process /WEN92/. Because of very strong smoke formation within adjacent containment compartments the wing was again closed at 14 min experimental time. However, closure was not completely efficient any more. Due to the thermal impact of the fire one door wing was deformed preventing its perfect reclosure. Consistent with the specification the exhaust blower from the second exhaust line, installed in the front of the fire compartment, received more power to increase the exhaust rate from 2500 m³/h to anticipated 8500 m³/h after 11 min experimental time. The supply of fresh air through the blower unit VB191 was cut off at 22 min experimental time.

Due to the actually observed development of the combustion process the fire-fighting system was activated at 14 min experimental time. Both events, the attempts to close the door again and the simultaneous activation of the fire-fighting system have not been communicated by the specification of the analytical task.

The load measurements of the racks I and II were assessed by the HDR-project in terms of "burning rates" and provided to the participants as input functions for their calculations.

In summary, the experiment may be divided into three distinguished phases:

**Phase 1:**
The evolution of the cable fire up to the moment, when the door of the fire compartment was opened at 8 min into the experiment (instead at 9 min as originally specified).

**Phase 2:**
The further development of the cable fire with fresh air supply from the containment by natural ventilation via the floor area in front of the fire compartment with the subsequent activation of the exhaust line connected to the floor compartment at 11 min up to the point in time (14 min) when the door was at least partially reclosed and the fire fighting system was activated.

**Phase 3:**
This period has not been covered by the provided specification of the experiment in the same manner as the earlier phases 1 and 2. It is characterized by getting the combustion
process under control with the fire fighting system in order reduce the consequences of the experiment for the entire containment (danger of dioxine-contamination). No "burning rate" has been communicated for points in time greater 14 min. This was considered as a matter of real "blind" predictions. The activation of the fire fighting system was also concealed.

With the necessary initial and operating conditions given to the participants they were requested to predict the thermal effects caused by the cable fire. A large number of sensors located within the fire compartment and the floor room in front of the fire compartment have been specified for comparison to calculated results. These parameters are listed in tables 1/1 to 1/3 which also give information about the elevation of individual sensors. Additionally, some parameters have been measured which characterize the conditions of the ventilation systems for both compartments actually linked to the containment ventilation system. All measured data have been locked by the HDR-project until the deadline for the submission of calculated results.

Temperatures of the atmosphere at various locations, the concentrations of combustion gases, heat transfer coefficients and heat flux to structures, mass flow rates of gases at various locations and the depth of hot gas layers in both compartments were selected as the main characteristic parameters describing the thermal consequences of the cable fire. Around 75 different parameters were specified.

The participants had the liberty to submit predictions either for the first two phases of the experiment up to 15 min only or for the anticipated entire duration of the cable fire up to 30 min. Results covering the third phase must be considered as a numerical benchmark exercise comparing analytical results amongst each other without meaningful comparison to the experimental background.

3 COMPARISON OF CALCULATIONS WITH MEASUREMENTS

A large number of experts actively working in the nuclear and non-nuclear fire protection area have been invited to utilize this unique offer to check the suitability of their analytical simulation models.

Eight institutions coming from 5 countries made an attempt to analyse the cable fire experiment E42.2 and submitted calculated results in response to the requirements of the task specification. Table 2 provides an overview on the participating institutions, their applied codes and the involved experts.
An identifier has been given to each participant for the preparation of the comparative plots. 5 participants applied multi-zone models, 1 participant a single-zone model and 1 participant employed a lumped parameter code with 3D-capabilities. Table 3 summarizes information received to be included within the comparison report. A variety of analytical simulation concepts has been utilized as evident from table 3.

The experiment involved three important areas, namely the fire compartment, the adjacent compartment separated by the door and the ventilation system. Accordingly, parameters requested to be predicted were separated into three groups associated to each area. The discussion of the comparison between the analyses and the experiment should be performed accordingly. Within the frame of this communication only a small number of comparative overlay plots can be shown and discussed. More plots will be published within the final comparison report of this exercise /KAR93/.

3.1 Predictions relevant for the Fire Compartment

As specified the cable fire was initiated by the gas burner located below channel 5 of rack I. The early development of the cable fire during phase 1 of the experiment has been monitored by temperature sensors CT5219 and CT5218, positioned at string B above and in the vicinity of racks I and II (see also fig. 5). Figs. 7 and 8 show the comparison between the measured temperature transient (signed EZMA) and the associated code predictions.

As evident from figures 7 and 8 the evolution of the combustion process during the first and second phase has been predicted with considerable scatter. A more accurate prediction has been expected because the mass deficits of racks I and II which are representative for the resulting combustion rates have been communicated to the participants by the specification. Hence, an appropriate transformation of combustion rates into thermal consequences should have yielded a more accurate prediction of the temperature evolution within those areas for which the evolution of the combustion process was in fact known through the mass deficit rates given by the task specification.

A similar observation can be made for the comparison of the measured and predicted temperature transients relevant for the location of the gas exhaust system (string A). Figs. 9 to 10 depict some predictions relevant for these locations. They differ only with respect to the elevation of the sensor. In particular for the first phase all code predictions differ by approx. 150-200 K from the measured transients, while during the second phase some predictions at least temporarily follow to the measured temperature transients with some time delay.
For the time later than 9 min, the specified constant exhaust gas flow rate from the fire compartment of 1700 m³/h was not met. Instead filter loading resulted in a continuous reduction as shown by fig. 11. This deviation caused the codes to calculate a higher fresh air inflow from the containment through the front room natural convection vent and through the fire room door into the fire room reducing calculated compartment temperatures.

During the third phase (after activation of fire fighting procedures) code calculations may be considered as a kind of numerical benchmark exercise. The results are indicative for the scatter of calculations in case neither mass loss rates nor pyrolysis rates of the involved cables are known in advance. A synopsis of the physical equations, the numerical solution procedure and the utilized code options deems necessary to shed more light into the existing uncertainty of cable fire predictions.

The comparison between measured and calculated gas temperatures at position D, 6.9 m within the door area (fig. 12) shows reasonable good predictions for the first phase while for the second phase the same group of codes shows more differences. The comparison of predictions and measurements for the lower position of string D, 4.85 m (fig. 13) which are representative for the gas temperature below the so-called "hot gas layer" shows again considerable deviations between the measurement and the variety of individual predictions. This is in particular true for the second phase and for later points in time.

One group of measured parameters was associated to the composition of the combustion gases within the fire compartment atmosphere. Most sensors were concentrated at position, string D. The deviations between measured and calculated results e.g. for the concentration of carbon dioxide (CO₂) are shown by fig. 14. They largely correspond to those observed for a number of local temperature predictions representative for the combustion.

Fig. 15 shows the predicted depth of the hot gas layer within the fire compartment. This is a characteristic parameter for several zone models. The scatter of predictions is selfspeaking. A representative measurement to compare to code predictions was obviously not available.

3.2 Predictions relevant for the Room in Front of the Fire Compartment and the Ventilation System

The room in front of the fire compartment became involved in the fire event at the moment the door was opened 8 min. into the experiment. The specification of the task indicated that the door was fully opened at 9 min. According to the protocol of the experiment /WEN92/
only 1 wing of the door was actually opened. This could also be considered as an important deviation from the technical specification of the experiment. In so far, all comparisons between measured and predicted parameters must be considered with a specific caveat.

Fig. 16 and 17 show temperatures within the front room, string E close to the door. They maybe considered as a linking information to the temperatures measured inside the fire compartment shown by figs. 12 and 13. Next to the position E information was available from position F showing again the temperature prediction for the upper section (fig. 18) and for the lower section of this compartment (fig. 19). These temperatures must be discussed together with the large deviations between measured and predicted velocities.

The fresh air supply to the front room is described by the local velocity sensor CF5382 which was equipped with a fan wheel anemometer. The comparison of the predictions with the measured transient is shown in fig. 20. At least one participant (AGB2) submitted a prediction which was very close to the measurement during the first phase of the transient. In the frame of this task measurements taken from the exhaust ventilation system provide an indication about the overall energy transport from the fire compartment to the environment via the filter components. In this sense, the measurements of the temperatures and gas velocities inside the exhaust ducts (250 mm diameter and 400 mm diameter) are of importance. Figs. 21 and 22 show a comparison between the measured and predicted gas velocities within both exhaust lines. Figs. 23 and 24 provide information about the measured and predicted temperatures for both gas exhaust ducts. Considerable deviations between measured and calculated velocities and temperatures are evident from the few submitted predictions.

4 OVERALL ASSESSMENT OF THE RESULTS

The modelling approaches and the overall results of this exercise have been discussed on occasion of a workshop held with the participants. An opportunity was given to evaluate in detail the reasons for the deviations between the "blind" predictions and the measured evolution of the parameters selected for the comparison.

Addressing the adequateness of the task specification a need for more exact information about some features of the experiments was identified (e.g. the time of opening the door, the activation of the fire fighting system). With respect to the activation of the fire fighting system it was indicated that only very few codes would have been able to simulate a fire
fighting process. Concerning the first two phases of the experiment the participants in general considered the specification as more or less adequate to evaluate the technical code application practise. With respect to the third phase of the exercise all participants agreed that a "benchmark-type" assessment of the few submitted calculations may confirm the impact of insufficient knowledge of pyrolysis rates for cable fire simulations.

However, a quantitative assessment of the consequences introduced into the computed results by a nonexhaustive specification may not become available without improved post-test calculations.

An important point for discussion is the interpretation of the provided weight loss curves, derived from weight measurements of the cable racks. The specification asked for the translation of cable weight loss rates into pyrolysis resp. burning rates yielding finally the relevant heat release rate to be applied for the thermohydraulic simulation of the containment internal atmospheric conditions. All experts were aware that the reliable prediction of the consequences of a cable fire presently would require the availability of a largely unknown cable weight loss rate as an input function.

It was commonly accepted opinion that the combustion of cables is a complicated thermo-chemical process with a strong feedback between generated temperatures and resulting further material disintegration (the pyrolysis process). No participating code would presently be capable to simulate the combustion process of cables in a relevant manner without providing at least the measured weight loss rate function.

A self-consistent model would be needed to more accurately predict the transient evolution of the generated thermal power and the resulting local temperatures.

As weight loss and pyrolysis rates are in practise not known in advance this standard problem exercise may also be considered as a "semi-blind" prediction of a fire inside the containment.

In summary, the standard problem exercise has been evaluated as very helpful to demonstrate the limited state-of-the-art in the simulation of a cable fire. A variety of codes has been employed. The exercise illuminated the achievable accuracy of "semi-blind" predictions for the experimental phases 1 and 2 under the condition that the thermal disintegration of the involved cables and the resulting weight loss rates are known in advance. Most comparisons to measurements may be considered as only partially satisfactory for both
phases. The submitted predictions for phase 3 are indicative. Reliable models are needed if
the simulation of a cable fire should also involve "blind" predictions of expected weight loss
rates and/or of pyrolysis rates of involved cables.

Some recommendations for supplementary research work have been derived. Items de-
serving future development work are:

- The pyrolysis process of cable isolation materials in particular and of solid combustible
  material in general

- improved nodalisation concepts satisfying basic requirements to simulate natural convec-
tion processes inside compartments and/or of interconnected compartments

- an improved understanding of the heat of combustion of involved cable materials and
  their reactants

- an improved method to determine combustion efficiency factors or effective stoichiometric
  factors

- for certain cases an improved simulation of the interaction between the filter loading pro-
cess and the achievable exhaust gas flow rates

All participants in the exercise expressed their appreciation to the Federal Ministry for Re-
search and Technology (BMFT), to the HDR-Project and to the Commission of the Euro-
pean Communities which all supported this activity.

References

/KAR92/ H. Karwat, K. Müller, U. Max
CEC Standard Problem "Prediction of Effects Caused by a Cable Fire Experi-
ment within the HDR-Containment

/KAR93/ H. Karwat
CEC Standard Problem "Prediction of Effects Caused by a Cable Fire Experi-
ment inside the HDR-Facility"
Final Comparison Report (to be published)

/MUL92/ K. Müller, H. Wegener, L. Löh
Auslegungsbericht: Brandversuche am HDR E42
Kabelbrand im geschlossenen System
PHDR Arbeitsbericht Nr. 40.031/91, Febr. 1992
/OEC89/ OECD-NEA
CSNI-Standard Problem Procedures
CSNI-Report No. 17, Rev. 3, November 1989

/SCH82a/ M. Schall
Design Report for the HDR-Containment Experiments V21.1 to V21.3 and V42
to V44 with Specifications for the Pre-Test Computations
PHDR Report No. 3.280/82, January 1982

/SCH92/ M. Schall, L. Valencia
Data Compilation on the HDR-Containment for Input Data Processing for Pre-
Test Calculations
PHDR Report No. 3.279/82, January 1982

/WEN92/ Wenzel, L. Lühr
Brandversuche am HDR; Kabelbrand im geschlossenen System
Versuche E42 (Versuchsprotokoll) 21.1-18.2.1992
PHDR Arbeitsbericht Nr. 40.042/92
Table 1/1: Parameters Specified for Comparison to Calculated Results

<table>
<thead>
<tr>
<th>Sensor No.</th>
<th>Parameter Identification</th>
<th>Physical Units</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fire Compartment</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CP 5201</td>
<td>Pressure fire compartment</td>
<td>mbar</td>
</tr>
<tr>
<td>CT 5219*</td>
<td>Temperature above racks I and II, Pos. B, 9.05 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5218</td>
<td>Temperature above racks I and II, Pos. B, 8.4 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5299*</td>
<td>Temperature near exhaust, Pos. A, 9.05 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5298</td>
<td>Temperature near exhaust, Pos. A, 8.4 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5297*</td>
<td>Temperature near exhaust, Pos. A, 7.4 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5296*</td>
<td>Temperature near exhaust, Pos. A, 6.1 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5295</td>
<td>Temperature cold gases near exhaust, Pos. A, 4.85 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5273*</td>
<td>Temperature, Pos. C, 6.25 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5276</td>
<td>Temperature, Pos. C, 7.4 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5285</td>
<td>Temperature hot gases door area, Pos. D, 6.9 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5280</td>
<td>Temperature cold gases door area, Pos. D, 4.85 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5031</td>
<td>Gas temperature inside Alsflex cover, channel 6</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5037</td>
<td>Gas temperature inside Alsflex cover, channel 14</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5069</td>
<td>Gas temperature inside Alsflex cover, channel 24</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5072</td>
<td>Gas temperature inside Alsflex cover, channel 25</td>
<td>°C</td>
</tr>
<tr>
<td>CG 5286</td>
<td>CO concentration door area, Pos. D, 6.4 m</td>
<td>Vol%</td>
</tr>
<tr>
<td>CG 5287</td>
<td>O₂ concentration door area, Pos. D, 6.4 m</td>
<td>Vol%</td>
</tr>
<tr>
<td>CG 5288</td>
<td>CO₂ concentration door area, Pos. D, 6.4 m</td>
<td>Vol%</td>
</tr>
<tr>
<td>CG 5289</td>
<td>CₙHₘ concentration door area, Pos. D, 6.4 m</td>
<td>ppm</td>
</tr>
<tr>
<td>CG 5280</td>
<td>CO₂ concentration cold gases, Pos. D, 4.85 m</td>
<td>Vol%</td>
</tr>
<tr>
<td>CV 5264</td>
<td>Hot gas velocity door area, Pos. D, 6.4 m</td>
<td>m/s</td>
</tr>
<tr>
<td>CV 5260</td>
<td>Cold gas velocity door area, Pos. D, 4.85 m</td>
<td>m/s</td>
</tr>
<tr>
<td>CF 1252</td>
<td>Depth of hot gas layer (not measured)</td>
<td>m</td>
</tr>
<tr>
<td>CQ 1952</td>
<td>Heat transfer coefficient (not measured)</td>
<td>W/m² K⁻¹</td>
</tr>
<tr>
<td>CQ 3952</td>
<td>Heat flux to structure (not measured)</td>
<td>W/m²</td>
</tr>
<tr>
<td>CM 1952</td>
<td>Mass flow of hot gases at door (not measured)</td>
<td>kg/s</td>
</tr>
<tr>
<td>CM 2952</td>
<td>Mass flow of cold gases at door (not measured)</td>
<td>kg/s</td>
</tr>
<tr>
<td>CE 5200</td>
<td>Thermal power of fire (calculated)</td>
<td>W</td>
</tr>
<tr>
<td>CE 2952</td>
<td>Integral heat input (calculated)</td>
<td>kJ</td>
</tr>
<tr>
<td>CA 5200</td>
<td>Pyrolysis rate (input function)</td>
<td>kg/s</td>
</tr>
</tbody>
</table>
Table 1/2: Parameters Specified for Comparison to Calculated Results (continued)

<table>
<thead>
<tr>
<th>Sensor No.</th>
<th>Parameter Identification</th>
<th>Physical Units</th>
</tr>
</thead>
<tbody>
<tr>
<td>Room in Front of the Fire Compartment</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CT 5310</td>
<td>Temperature, Pos. F, 9.1 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5318</td>
<td>Temperature, Pos. G, 9.1 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5315</td>
<td>Temperature, Pos. G, 7.3 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5314</td>
<td>Temperature, Pos. G, 4.85 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5323</td>
<td>Hot gas temperature, Pos. H, 8.6 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5320</td>
<td>Temperature, Pos. H, 4.85 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5328*</td>
<td>Temperature, Pos. J, 8.6 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5325*</td>
<td>Temperature, Pos. J, 4.85 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5303</td>
<td>Temperature, Pos. E, 7.5 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5301*</td>
<td>Temperature of cold gases, Pos. E, 4.85 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5309</td>
<td>Temperature, Pos. F, 7.5 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5305</td>
<td>Temperature of cold gases, Pos. F, 4.85 m</td>
<td>°C</td>
</tr>
<tr>
<td>CV 5310</td>
<td>Velocity of hot gas, Pos. F, 9.1 m</td>
<td>m/s</td>
</tr>
<tr>
<td>CT 5384*</td>
<td>Temperature, Pos. K, 9.1 m</td>
<td>°C</td>
</tr>
<tr>
<td>CT 5382</td>
<td>Temperature, Pos. K, 4.85 m</td>
<td>°C</td>
</tr>
<tr>
<td>CV 5305</td>
<td>Velocity of cold gas, Pos. F, 4.85 m</td>
<td>m/s</td>
</tr>
<tr>
<td>CF 5382</td>
<td>Velocity of cold gas, Pos. K, 4.85 m</td>
<td>m/s</td>
</tr>
<tr>
<td>CG 5310</td>
<td>CO₂ gas concentration hot gases, Pos. F, 9.1 m</td>
<td>Vol%</td>
</tr>
<tr>
<td>CG 5305</td>
<td>CO₂ gas concentration hot gases, Pos. F, 4.85 m</td>
<td>Vol%</td>
</tr>
<tr>
<td>CF 1253</td>
<td>Depth of hot gas layer (not measured)</td>
<td>m</td>
</tr>
<tr>
<td>CF 3253</td>
<td>Neutral plain height (not measured)</td>
<td>m</td>
</tr>
<tr>
<td>CM 2954</td>
<td>Mass flow of cold gases (not measured)</td>
<td>kg/s</td>
</tr>
<tr>
<td>CM 1953</td>
<td>Mass flow of cold gases (not measured)</td>
<td>kg/s</td>
</tr>
<tr>
<td>CM 2953</td>
<td>Mass flow of cold gases (not measured)</td>
<td>kg/s</td>
</tr>
<tr>
<td>CE 3953</td>
<td>Heat flux to structure (not measured)</td>
<td>W/m²</td>
</tr>
<tr>
<td>CE 1953</td>
<td>Enthalpy flow hot gases (not measured)</td>
<td>kJ/s</td>
</tr>
<tr>
<td>CE 1453</td>
<td>Enthalpy flow cold gases (not measured)</td>
<td>kJ/s</td>
</tr>
</tbody>
</table>
Table 1/3: Parameters Specified for Comparison to Calculated Results (continued)

<table>
<thead>
<tr>
<th>Sensor No.</th>
<th>Parameter Identification</th>
<th>Physical Units</th>
</tr>
</thead>
<tbody>
<tr>
<td>CT 1101</td>
<td>Temperature air supply duct</td>
<td>°C</td>
</tr>
<tr>
<td>CT 6212</td>
<td>Temperature exhaust duct 250 mm ø</td>
<td>°C</td>
</tr>
<tr>
<td>CT 6650</td>
<td>Temperature exhaust duct 400 mm ø</td>
<td>°C</td>
</tr>
<tr>
<td>CG 6286</td>
<td>CO gas concentration exhaust duct 250 mm ø</td>
<td>Vol%</td>
</tr>
<tr>
<td>CG 6287</td>
<td>O₂ gas concentration exhaust duct 250 mm ø</td>
<td>Vol%</td>
</tr>
<tr>
<td>CG 6288</td>
<td>CO₂ gas concentration exhaust duct 250 mm ø</td>
<td>Vol%</td>
</tr>
<tr>
<td>CG 6289</td>
<td>C₆H₆ gas concentration exhaust duct 250 mm ø</td>
<td>ppm</td>
</tr>
<tr>
<td>CV 6211</td>
<td>Gas velocity exhaust duct 250 mm ø (Prandtl-Tube)</td>
<td>m/s</td>
</tr>
<tr>
<td>CF 6212</td>
<td>Gas velocity exhaust duct 250 mm ø (from fire comp.)</td>
<td>m/s</td>
</tr>
<tr>
<td>CF 6650</td>
<td>Gas velocity exhaust duct 400 mm ø</td>
<td>m/s</td>
</tr>
<tr>
<td>CM 1911</td>
<td>Mass flow air supply calculated</td>
<td>kg/s</td>
</tr>
<tr>
<td>CM 1912</td>
<td>Mass flow exhaust duct 250 mm ø calculated</td>
<td>kg/s</td>
</tr>
<tr>
<td>CE 1911</td>
<td>Enthalpy flow exhaust duct 250 mm ø calculated</td>
<td>kW/s</td>
</tr>
<tr>
<td>CE 1912</td>
<td>Enthalpy flow exhaust duct 400 mm ø calculated</td>
<td>kW/s</td>
</tr>
<tr>
<td>Country</td>
<td>Institution</td>
<td>Experts</td>
</tr>
<tr>
<td>---------</td>
<td>-------------</td>
<td>---------</td>
</tr>
<tr>
<td>Austria</td>
<td>Arbeitsgemeinschaft Brandsicherheit München/Wien</td>
<td>Prof. Dr. Schneider Mr. Lebeda</td>
</tr>
<tr>
<td>Finland</td>
<td>Technical Research Centre of Finland</td>
<td>Dr. O. Keski-Rahkonen</td>
</tr>
<tr>
<td>France</td>
<td>Electricité de France</td>
<td>Mr. E. Chabert</td>
</tr>
<tr>
<td>Germany</td>
<td>Battelle Europe</td>
<td>Prof. Wolf</td>
</tr>
<tr>
<td></td>
<td>Ingenieurgesellschaft für Technische Sicherheit</td>
<td>Mrs. Büttner</td>
</tr>
<tr>
<td></td>
<td>Siemens AG, KWU</td>
<td>Mr. Samman</td>
</tr>
<tr>
<td></td>
<td>TU Braunschweig</td>
<td>Prof. Hosser, Dr. Heins</td>
</tr>
<tr>
<td>USA</td>
<td>Sandia National Laboratory</td>
<td>Dr. V. Nicolette</td>
</tr>
</tbody>
</table>

Table 2: Institutions Submitting "Blind" Post-Test Analyses of the HDR-Cable Fire Experiment E42.2
<table>
<thead>
<tr>
<th>Institution</th>
<th>Code</th>
<th>Fire Compartment</th>
<th>Front Room</th>
<th>Ventilation</th>
<th>Total</th>
<th>Flow Junctions</th>
<th>Heat Structures modelled?</th>
</tr>
</thead>
<tbody>
<tr>
<td>AGB2</td>
<td>MRFC</td>
<td>1</td>
<td>2</td>
<td>2</td>
<td>5(+1C)</td>
<td>8 ?</td>
<td></td>
</tr>
<tr>
<td>VTT</td>
<td>BRI2</td>
<td>2</td>
<td>2</td>
<td></td>
<td>4</td>
<td>4</td>
<td>yes</td>
</tr>
<tr>
<td>EDF</td>
<td>MAGIC</td>
<td>2</td>
<td>2</td>
<td>pressure dependent</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>BF</td>
<td>GOTHIC</td>
<td>4</td>
<td>4</td>
<td>3</td>
<td>11 (+15C)</td>
<td>53</td>
<td>69</td>
</tr>
<tr>
<td>IGT</td>
<td>FAST</td>
<td>2</td>
<td>2</td>
<td></td>
<td>4 (+ Env.)</td>
<td>9</td>
<td>yes</td>
</tr>
<tr>
<td>KWU</td>
<td>TEMPW</td>
<td>1 (4)</td>
<td>2 (8)</td>
<td></td>
<td>3 (+ 8C)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TUB</td>
<td>FIGARO</td>
<td>1</td>
<td>2</td>
<td></td>
<td>10</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SNL</td>
<td>COMPBRN</td>
<td>1</td>
<td>0</td>
<td>2</td>
<td>1</td>
<td>0</td>
<td>yes</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th><strong>Table 3:</strong> Main Features of the Analytical Simulation Models (ASM)</th>
</tr>
</thead>
<tbody>
<tr>
<td>(C= outside fire and front room compartment)</td>
</tr>
</tbody>
</table>
Fig. 1: Vertical Cross Section of the Containment (General View)
Fig. 2: Horizontal Cross Section of Level 1.500 of the HDR Containment
Fig. 4: Scheme of the Ventilation System for the Fire Compartment
Fig. 5: Main Position of Sensor Strings Arrangement within Fire Compartment and Front Room
Fig. 6: Arrangement of Cable Trays on Racks I and II
Fig. 7: Temperatures above Cable Racks (String B, 9.05 m)
Fig. 8: Temperatures above Cable Racks (String B, 8.4 m)
Fig. 9: Temperatures near Gas Exhaust (String A, 9.05 m)
Fig. 10: Temperatures near Gas Exhaust (String A, 8.4 m)
Fig. 11: Velocity of Gas actually exhausted from the Fire Compartment through 250 mm ø Line
Fig. 12: Temperature of Hot Gas, Door Area (String D, 9.6 m)
Fig. 13: Temperature of Cold Gas, Door Area (String D, 4.8 m)
Fig. 14: CO₂-Concentration, Door Area (String D, 6.4 m)
Fig. 15: Depth of Calculated Hot Gas Layer (not measured)
Fig. 16: Temperature upper Front Room close to Door (String E, 7.5 m)
Fig.17: Temperature lower Front Room close to Door (String E, 4.8 m)
Fig. 18: Temperature Ceiling Front Room (String F, 9.1 m)
Fig. 19: Temperature Bottom Front Room (String F, 4.8 m)
Fig. 20: Velocity of Fresh Air Supply (String K, 4.8 m)
Fig. 21: Velocity of Gas Exhaust from Fire Compartment (250 mm Ø Line)
Fig. 22: Velocity of Gas Exhaust from Mounting Hatch (Front Room, 400 mm ø Line)
Fig. 23: Gas Exhaust Temperature 250 mm ø Line
Fig. 24: Gas Exhaust Temperature 400 mm ø Line

33°C
Contribution to PWG No. 1 Specialist Meeting  
Fires and Fire Protection Systems in NPPs  

Special Effects Studies During HDR Fire Experiments And Their Calculability

K. Müller, Kernforschungszentrum Karlsruhe  
R. Dobbernack, Technische Universität Braunschweig  
U. Max, Ingenieurbüro

Abstract

Since 1984 fire experiments have been run at HDR. They included different fire sources as well as differing geometries from one experiment to the other. Prediction of fire behavior was one goal of these research activities, calculability of the effects the other one. As a side product punctual answers were gained concerning special effects not yet known very well because of the complexity of a nuclear power plant as:
- Temperature distribution and mixing effects in safety relevant regions near the fire area.
- Soot formation/transport and depletion behavior throughout the plant and their influence on plant filter clogging.
- Behavior of pressurized rescue stairways and prolongation of accessibility.
- Insulation effectiveness of components.
- Formation and distribution of toxic components during cable fire.
- Opening of doors on fire propagation
- Fire fighting by spray systems.

While temperature distribution and mixing is partly predictable by calculation, soot formation and depletion leads to serious non predictable effects inside the plant, which need plant specific preparations. Filter clogging seems to be such a specific problem. Rescue and fire attacks needs space for the involved personal to organize their approach to the fire site. The results we have found concerning visibility do not cooperate with today fire fighting instructions. In combination with opening doors to fire areas, there are still serious deficiencies between practice and theory. During cable fire, the effectiveness of insulation and the distribution of toxic products have been measured. Nevertheless the overall behavior of a NPP against fire seems to be rigid and without safety problems compared with other accident scenarios.

Description of the Experiments

HDR, a decommissioned test reactor located in the Federal Republic of Germany, has been used for reactor safety experiments since 1975. It is available for fire tests too since 1984. The facility with about ~ 11 000 m³ free volume, partitioned into about 62 compartments, equipped with components, pipework, concrete walls, etc., is a great step towards reality within the framework of integral fire experiments conducted at an international level. (Fig. 1)

Experiments were planned with gas, wood, oil and cables as combustibles. Fig. 2 shows the overall program between 1984 and 1991. The first tests starting with gas and wood crib fires were to study the behavior of the facility at low energy output rates.
Three fire areas where chosen during the planning. (Fig. 3)

1. Low plane
   including the most of the plant into fire induced temperatures profiles.
2. High plane
   for simulation of a cable fire near the dome area.
3. Mid plane
   for simulation of an oil fire in the pump area.

First experimental series including forced ventilation gas fires and natural ventilated wood crib fires were to learn about plant behavior in general, to test the containment behavior under different ventilation conditions and to enlarge the measurement experience.

Next experiments were performed in the HDR near the containment dome. The aim of these experiments was to simulate fire of cable trays near the dome and to verify fire codes calculating the main physical parameters only knowing the burning rate of the fire. As cable fires produce a lot of corrosive gases an oil fire source was chosen to protect the HDR containment.

During the fire 300 measuring points registered the temperatures in the fire room, the temperatures, velocity and gas concentration (O₂, CO₂, CO) in the opening of the fire room and in the 19 m height plume and the convection pattern throughout the whole containment.

The overall plant behavior encouraged to perform large pool fires in a mid plane of the plant. Besides the scientific goals a couple of technical questions were included as plant filter, rescue area behavior and aerosol production.

Fig. 3 gives a representation of the fire compartment and the fire level. Fuel is burned in pools of different sizes standing on a balance. The total amount of combustible per test is stored in the pool burning in an uncontrolled manner after ignition. An isoparaffin hydrocarbon mixture in the C₁₁ - C₁₂ range classed as "low soot" grade was used as combustible.

The size of the door leading to the fire compartment plays a decisive role in fire development. Air is supplied and smoke is evacuated through this door and further via the natural passages in the building, such as staircases, installation wells and connections between the compartments.

The proposal to prolongate rescue times of personal was evaluated with a pressurized staircase construction. The elevator shaft of the plant was prepared for this purpose.

The following oil pool tests were to get knowledge about different parameters influencing the fire propagation. The fire mid plane was used again. A possibility to close and open the compartment doors was installed. Pan size and oil feeding could be varied. Two pan materials were used.

The last experimental series was performed with PVC-cables inside the same compartment to get comparable results.

The floor in front of the fire chamber was protected by circular walls and a closed ceiling construction to avoid contamination by dioxin containing aerosols inside the HDR plant. The volume of the floor is 108 m³, the surface areas are ~ 160 m². A fresh air opening above the ground ensures natural ventilation to the fire compartment when connected to the floor region. A local ventilation system was connected to the floor and fire chamber to permit forced ventilation fires and the additional air intake will cause natural ventilation to the local area.

The exhaust lines from fire room and floor were connected to a cleaning system within the containment. Fig. 4

The cleaning line between the fire area and the HDR ventilation consists of a jet pipe cleaner (VB104) and a wire mesh filter unit (VB191) where the smoke and aerosols as well as the gaseous hydrochloric acid from the burning cables are washed out.
In the fire compartment three horizontal cable racks are installed containing 20 cable channels of 20 cm width and 3.5/4.5 m length. Three additional channels of 30 cm width and 4.5 m length are vertically fixed to the wall. All cables were installed at the beginning of the experiments but heavily insulated with Aisiflex matting when not involved in the fire. The cables were chosen from a conventional plant installation. They comprised power cables as well as signal cables of varying diameter between 9 -23 mm with PVC-cover. The amount of cabling is indicated on each cable channel.

**Results of the Experiments of the Gas and Wood Crib Fires**

The results showed:
- a linear behavior of the plant concerning loads
- heavy mixing of gases reducing the temperature attack
- deficiency of code results to measurements due to "state of the art knowledge of codes" used for calculation of these problems.

The results of the high plane fires are:

In general all fire room temperatures are quasi-stationary after 10 minutes fire duration due to the constant oil supply. Maximum of 1500°C are found in top of the fire room. During the quasi-stationary phase temperatures of 1000°C were measured. The temperatures near the floor of the fire room are influenced by incoming cold air. In the door opening temperatures of 900 - 1300°C were measured in the hot gas layer. The fire turned from a fuel controlled to a ventilation controlled fire after 2 minutes.

In a short distance behind the fire room two measure grids and single measurement points were installed in the plume. So data about temperature-, velocity- and gas-concentration profiles are available. The temperature distribution in the two grids show after 6 minutes fire duration the influence of mixing effects between hot gas and the surrounding air. The cooling of the hot gas is very strong on its 19 m way towards the steel shell. (Figs. 7/8)

Short after ignition a thermal induced gas circulation caused an uniform distribution of gas concentration and soot in the whole containment. The vent air filter and the recirculating air filter used for cleaning the containment air lost their function after short time. Visibility than dropped to zero, so that even personnel knowing the plant layout will hardly be able to identify and fight the fire source. The soot spread over the whole plant, even in closed rooms. Deposition of soot was found on all horizontal surfaces (1 - 10 mm thick). As the CO₂-concentration increased to values of 3% after a 30 min fire duration, no fire will stop in want of oxygen in a real nuclear power plant for a long time.

The cable experiments covered the range of results between laboratory scale and worst case fire, depending on localisation of ignition in the hot or cold layer of the fire chamber. Two of the planned experiments were laboratory scale with moderate fire propagation rates, low room temperature and no attack to neighbouring trays. Opening of fire door did not influence fire dynamic.

In the experiment E42.2 nearly all remaining cables trays, insulated or non insulated went on fire within minutes and heavy sooted gases escaped from the fire chamber. In this case the opening of the door ignited the burnable gases in the whole room and in the separated channels underneath insulation and fire fighting was necessary to stop further fire propagation.
Transfer of results to real NPP's

The HDR results can be evaluated concerning transferability to real plants. In detail the following topics can be discussed.

- Prediction of fire development in closed areas
- behavior of filter systems
- behavior of rescue areas during fire
- smoke convection pattern inside a plant.
- aerosol production and depletion
- function of fire protection measures as isolation
  function of stationary fire fighting equipment
  function of insulation material
- behavior of doors
- steel shell as a heat sink
- protection of personal and escape times via pressurized stairways
- distribution of smoke and aerosols throughout the plant
- visibility decrease in the plant
- temperature attack to components and to the outer shell
- possibility of secondary ignitions
- aerosol and temperature attack to safety related equipment
- plant main filter attack by temperature and soot
- contamination with toxic products during cable fire.

Tests have been carried out from small scale up to reality scale and worst-case fires. General experimental solutions found are:

- The fresh air volume inside a plant will always be sufficient for the amount of combustible provided.

- Combustion rates of combustible (oil, wood, cable) were clearly higher than measured in open air tests due to interconnection between room walls and fire source. After short unexpected high temperatures prevailed in the fire compartment.

- Local temperatures may cause serious destructions of the individual cells of the concrete structure and of components and they may give rise to secondary ignitions, in the near field zone of a fire if there is any burnable materials as cables or electronic equipment. The main destruction will occur from radiative heat.

- Measured fluctuations in the combustion rate are nearly not reflected in the temperature plots traced in the far field.

- Heavy mixing of cold air and fume gases will bring temperatures down within short distance from the fire chamber, the dome temperature never exceeded crucial values in safety terms

A division into a "near field" and "far field" areas of the fire was confirmed also for high power inputs of 8 - 10 000 kW.

- Smoke and toxic gases will be distributed within the plan thin short time using the natural convection loops available as staircases, open ventilation ducts and room connections.

- Visibility will be reduced to zero within minutes. Fig. 9

- Temperatures of max. 120 C were measured in the vent air ducts, the differential pressure loading of the plant exhaust filters remained comparatively low.

- Pressurized stairways for rescue and free fighting access remain nearly untouched, when being vented with sufficient air changes Fig. 10
Soot production is highly dependable on fire propagation and burning status. A ventilation controlled fire will cause high soot distribution and depletion throughout the plant.

In an open cable fire containing PVC the soot will be highly contaminated by Dioxins and Furans and will lead to expensive cleaning action concerning personal protection besides the remaining nuclear aspects. High soot amounts will rise the differential pressure of nuclear prefilters within short times to the limits. Main filters will remain unclogged because of their surface areas remaining comparatively open during dry soot loading.

A coarse wire mesh filter unit with a separate ventilator running inside the plant will catch soot particles from the air and will help to hold visibility for longer times.

Lessons learned from the ongoing data evaluation are:
Cable fire in lower parts of compartment will not attack the surroundings and only cause small damage and low temperatures. Cable fire taking place in upper regions of cable channels will turn violent when opening a door, thus giving chance to new oxygen coming into contact with the gasesous products of cable deterioration process. This gas mixture will even ignite other cables within protective barriers when being warmed up before. There are indications that fire fighting will make things worse by wetting protective insulation thus changing the heat transfer coefficients of the insulation material.

The further evaluation of experimental data will hopefully contribute to future fire protection actions inside Nuclear Power Plants.

Calculability of Experiments

The fire experiments have been accompanied by extensive calculations with different fire codes. They included zone models, flow-models and field models (Fig.11). Calculations have been carried out with and without knowledge of the burning rate including different geometries inside the plant and different combustibles as gas, wood cribs, oil and cables. Different fuel sources as premixed flames, oil pool or constant oil feeding; solids (wood, cable) were taken into account. The results show the state of the art of codes at the start of the HDR fire research program and the increasing knowledge of codes when following the experimental evaluation of fire phenomena.

The code verification ended in a CEC benchmark exercise on "cable fire".
Fig. 1: HDR longitudinal cut
<table>
<thead>
<tr>
<th>Tests</th>
<th>Material</th>
<th>Level</th>
<th>Dates</th>
</tr>
</thead>
<tbody>
<tr>
<td>6 Tests</td>
<td>Gas 1</td>
<td>Level 1.400</td>
<td>T51.11-19 '84/85</td>
</tr>
<tr>
<td>3 Tests</td>
<td>Wood cribs</td>
<td></td>
<td></td>
</tr>
<tr>
<td>5 Tests</td>
<td>Gas 2</td>
<td>Level 1.900</td>
<td>T51.21-25 '86</td>
</tr>
<tr>
<td>4 Tests</td>
<td>Hydro carbon open</td>
<td></td>
<td>T52.11-15 '87</td>
</tr>
<tr>
<td>4 Tests</td>
<td>Hydro carbon open</td>
<td></td>
<td>E41.1-4 '88</td>
</tr>
<tr>
<td>6 Tests</td>
<td>Hydro carbon closed</td>
<td></td>
<td>E41.5-10 '90</td>
</tr>
<tr>
<td>3 Tests</td>
<td>Cable fire closed</td>
<td></td>
<td>E42 '92</td>
</tr>
</tbody>
</table>

Fig.2: Experimental fire programme
Fig. 3: Fire Room Geometries

Experiment No. | Energie
---|---
T 52.11-15 | 2-4000 kW
Oilpool

E41.1-4 | 6-8000 kW
Oilpool open

E41.5-10 | 1-4000 kW
Oilpool and feeding closed compartment

E42 | 0,5-2500 kW
cable closed compartment

T51.11-14/19 | 200-1000 kW
gas, forced ventilated

T51.16-19 | 1-2500 kW
woodchips natural ventilated

T51.21-25 | 1000 kW
gas, with ventilation duct
Fig.5: Temp. distribution along fire level
GRID I (39M)
MESH WIDTH (1M)

GRID II (31M)
MESH WIDTH (0.5M)

Fig. 7: Temp. profil along plume t = 6min

343
Fig. 8: Temperature profile of the plume and steel shell temperature
Fig. 9: Visibility in the Plant
Fig. 10: CO2 - VOL% E41.1/3
Session #5

Risk Assessment

Mr. Zhong, Chairman
FIRE PSA TVO I/II

Kari Taivainen
Eero Nikula
Risto Himanen

Teollisuuden Voima Oy
Olkiluoto, Finland

Olkiluoto, November, 1993
ABSTRACT

Fire PSA TVO I/II is one part of the level 1 living PSA of the nuclear power plant TVO I/II. The other two parts of the living PSA model are floods and internal initiators. The event trees and fault trees including the reliability data attached to them are updated constantly so that the PSA model responds to the current status of the plant. At the moment the contribution of the internal fires to the annual core damage risk is 25%. The contribution of the internal floods is slightly smaller, 15%. The rest of the core damage risk is induced by the internal initiating events (LOCAs and transients, 40%) and the refuelling period (20%).

Initially, Fire PSA included all the important safety related equipment and their power supplies. Recently, the scope of the fire analysis has been extended by taking into account the important instrumentation systems, too. This extension has been implemented by mapping out the equipment and the cables of the reactor protection system (RPS) and the controls of the safety related equipment. The consequences of the instrumentation system fires include spurious trippings and spurious control orders in addition to the normal failure modes like “failure to start” or “inadvertent stop”.

This year there has been made a number of significant improvements both in the active and passive fire protection of the plant. Due to these plant modifications the core damage frequency induced by the fire initiators has been reduced from the value 2·10⁻⁵/year to 4·10⁻⁶/year, i.e. by factor 5.
1 INTRODUCTION

Teollisuuden Voima Oy (TVO) owns and operates two identical 710 MW ABB Atom type BWR units, TVO I and TVO II, in Olkiluoto (Finland). The units have been in commercial use since 1979 and 1982, respectively.

TVO I/II PSA program was initiated in 1984 by the utility [1]. The level 1 analysis was originally performed for internal initiating events, but later it was extended to include also flood and fire initiators. Floods and fires were considered to be the most important external events at the plant specific conditions [2]. They occur frequently from the statistical point of view and they have significant impact on the plant safety. Flood and fire analysis were also required by the Finnish regulatory body.

The results of TVO I/II PSA indicate that the contribution of the fires to the total core damage risk is 25 % and the contribution of floods 15 %. In addition to floods and fires the total core damage risk consists of the internal initiating events (LOCAs and transients). All these three analysis have been performed using slightly different methods and assumptions. Therefore, the comparison of results has to be done very carefully. The largest uncertainties compared with each other lie in the determination of initiating event frequencies. System and plant level models in flood and fire analysis have been implemented by modifying fault and event trees used in the analysis of internal initiators. Only CCI (Common Cause Initiator) type fires are included in the living PSA.

The probabilistic safety assessment of other external events than floods and fires has not been considered necessary up to date. Many potential external events can be ignored due to the location and the environment of the plant units. For example, the seismic phenomena have been ignored because of the very old and stable bedrock.

The principle of using four half capacity subsystems is applied in both units to all important safety functions. The four redundant circuits of the safety related systems are assigned to two main groups, located in physically separated areas. Within these areas, separation by distance or by means of fire barriers is used between the redundant circuits.

The fire safety design criteria for the TVO plants are based on Swedish codes and standards. Adaptations have been made to Finnish regulations. The plant has very wide active and passive fire protection systems and also an on-site fire brigade.

2 SCOPE OF FIRE PSA TVO I/II

Fire PSA of TVO I/II was reported for the first time to the Finnish regulatory body in 1991. At that time the impact of fire events on the safety related equipment and its power feeding had been analysed [3]. In 1993 Fire PSA has been extended to include the important safety related instrumentation systems too. The Shut Down Fire PSA has been planned to be carried out within three years.

The external fires have been excluded in Fire PSA due to their low occurrence probability. The plant units are surrounded by the sea on the one side and the large area of bare land on the other sides. The possibility of losing connection to external electric net due to a forest or a
grass fire has been taken into account in the analysis of the internal initiating event "Loss of external grid". A grass fire under the external grid connecting power lines has once tripped one plant unit. However, the consequences of the event do not differ from the ordinary loss of external grid. Thus Fire PSA deals only with the internal fire events.

3 DETERMINATION OF FIRE FREQUENCIES

The room specific conditional fire frequencies needed for the calculation of the fire initiator frequencies have been determined using Berry's method [3, 4]. According to Berry's model, a fire can start only if there exists combustible material in the room and it gets into contact with an ignition source. The persons present in the room and the mechanical and the electrical equipment are potential ignition sources. In the case the fire load and an ignition source get into contact with each other the fire ignites with the probability which depends on the ignition temperature of the combustible material.

Berry's model also takes into account the possibility that the initial fire can be detected and suppressed by the persons present in the room or it can be extinguished itself. The fire detection and suppression probabilities depend on the presence of persons and the type of combustible material. The self extinguishment probability depends on the distribution of the fire load.

The room specific probabilities needed in the Berry's model have been obtained by visiting all the rooms at the plant and making expert judgement. When the ignition probabilities have been calculated for each room, the total fire frequency gained from the statistics can be allocated to the single rooms in the proportion of the ignition probabilities. The total fire frequency estimate of the plant units is based on the experience of TVO I/II and the same type units in Sweden.

4 FIRE PROPAGATION

One major conservative assumption has been made: if there is a fire in a room, it will make all active components in that room unavailable. Later, there has been made some more detailed analysis concerning fire propagation in many cable rooms provided with automatic sprinkler systems. Fire propagation has also been analysed in relay rooms and in the pump rooms where the pumps of service water system and several component cooling water systems are located.

The assumption, that a fire will destroy all the equipment in a room, will give a very conservative estimate about the initiating event frequencies. Therefore, three different fire propagation parameters have been used in Fire PSA. The parameter $p_1$ describes the fire propagation from one redundant subsystem to another. When the cables of two subsystems are located in a same runway they are separated by distance and automatic sprinkler systems. This physical separation has been taken into account quantitatively by using conditional probabilities ($p_1$). The second parameter $p_2$ describes the fire propagation inside one electric subsystem. This parameter has been used mainly for reducing the fire propagation probability from upper located power cables to better sheltered signal cables. The third parameter $p_3$ tells the failure mode of the instrumentation cables.
5 PLANT RESPONSE ANALYSIS

The classification of fire events is based on the very careful room by room mapping out of the safety related components. During the first phase of Fire PSA a database was formed of the safety related active components and their power supplies. The essential part of the work was the mapping out of cable routes. At the first phase only the power cables of the equipment were taken into account. At the second phase the important instrumentation systems were added to the database too. This extension included two major parts of instrumentation, the reactor protection system (RPS) and the controls of the safety related equipment.

The cable routes were stored as a database which can be used for the room by room search. This will make it possible to search for a single room in the database and see which equipment and cables are located in that room. This information has been combined with the fire propagation parameters to find out the contribution of the event to the fire initiator.

Room fires were divided to initiating event groups by investigating the plant response of each fire. Two different cable failure modes were considered as a result of the cable fire: short circuit or open circuit. In the case of the power cable fire it can be assumed that the power supply to the component will be interrupted. Fires in the cables of reactor protection system can cause spurious trippings that may prevent the function of some safety related systems or subsystems. Fires in the control cables of the equipment can cause spurious starts/stops of pumps or closings/openings of the valves, or prevent controlling of some components from the main control room.

The frequencies of the initiators were calculated by summing the fire frequencies of the single rooms in an initiating event group and taking into account the fire propagation parameters. A total of 15 event trees have been constructed according to the classification of fires. Most event trees are based on the event tree of the internal initiating event analysis where the main feed water system has been lost. The influence of fires on the total core melt frequency can be estimated by quantifying the event tree analysis.

6 RESULTS AND PLANT MODIFICATIONS

The results of Fire PSA indicate that the largest contributors to the core damage risk are fires in such cable rooms where two redundant subsystems are present, and fires in pump rooms where the pumps of service water system and several component cooling water systems are located. In the rooms containing cables of two subsystems it is possible to lose two trains of many DC systems and get spurious trippings from the 2/4-coupled reactor protection conditions. The pump room fires contribute to the core damage risk mostly through the dependencies that affect many emergency cooling systems.

On the basis of Fire PSA and deterministic fire safety thinking fire protection improvements have been planned and are now being carried out. These actions include improvements in the sprinkler systems of the cable rooms and in the passive fire protections of the cables. The number of sprinkler heads in existing distribution pipes has been increased and some heads have been moved to the locations where they can properly work. Many sprinkler heads have been changed to quick response sprinklers. Sprinkler systems have also been installed to new areas.
In the pump rooms the power feeding cables of the service water pumps and cooling water pumps have been protected with the fire retardant cable coating between the pump motors and the wall. This action prevents or slows down the fire propagation from the pump motor to the junction where the cables of many pumps cross. The passive fire protection of the cables has also been improved in two cable spreading rooms of the control building where most cables to the AC/DC converters are fed. Two large rooms containing cables of two trains have been divided to fire departments containing only cables of one train. One crossing horizontal cable tray group of the other train has been covered completely with the fire retardant coating to make separation as good as possible.

Before accomplishing the fire protection improvements, the core damage frequency induced by the fire initiators was estimated to be $2 \times 10^{-5}$/year. When the improvements are completed by the end of the year 1993, the core damage frequency will be reduced to $4 \times 10^{-6}$/year, i.e. by factor 5.

Probabilistic fire analysis has been proved to be a useful tool in risk management. It gives an alternative view to the deterministic calculations and thinking when comparing corrective actions or improvements.

REFERENCES


TVO I/II PSA RESEARCH PROGRAM

<table>
<thead>
<tr>
<th>Project</th>
<th>84</th>
<th>85</th>
<th>86</th>
<th>87</th>
<th>88</th>
<th>89</th>
<th>90</th>
<th>91</th>
<th>92</th>
<th>93</th>
<th>94</th>
<th>95</th>
<th>96</th>
<th>97</th>
<th>98</th>
</tr>
</thead>
<tbody>
<tr>
<td>DECISION &amp; REQMTN</td>
<td></td>
<td></td>
<td>1</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>PILOT STUDY</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>BASIC PSA (LVL 1, INT)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>FIRES</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>FLOODS</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SHUT DOWN EVENTS</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LIVING, PLANNING</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LIVING, IMPLEMENT.</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LVL 2, PLANNING</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LVL 2, PRESTUDY</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LVL 2, STUDY</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LVL 2, LIVING</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

27.8.93SCHEDULE.XLS/RPH
Core damage frequency distribution according to IE causes (2E-5/y)
COURSE-CONSEQUENCE DIAGRAM OF BERRY'S MODEL

Persons

Mechanical equipment

Electrical equipment

Contact with combustible material

A

Does it ignite?

no

1-B

B

Has it been found?

yes

Does it extinguish?

yes

C2

no

1-C2

1-C1

no

Pilot fire

Is it dying?

yes

F

no

1-F

FIRE

NO FIRE
**Mapping Out Form for Rooms**

<table>
<thead>
<tr>
<th>TVO/Fire-PSA</th>
<th>Room code</th>
</tr>
</thead>
<tbody>
<tr>
<td>19</td>
<td></td>
</tr>
<tr>
<td>Mapping out form for rooms:</td>
<td>Similar to:</td>
</tr>
</tbody>
</table>

**Radiation Classification**

<table>
<thead>
<tr>
<th>Radiation</th>
<th>Surface contaminat.</th>
<th>Air contamination</th>
</tr>
</thead>
<tbody>
<tr>
<td>r o g</td>
<td>r o g</td>
<td></td>
</tr>
</tbody>
</table>

**Persons in the Room (A1, C1)**

<table>
<thead>
<tr>
<th>Always</th>
<th>Mostly</th>
<th>1/3 time</th>
<th>On rounds</th>
<th>Seldom</th>
<th>5</th>
</tr>
</thead>
</table>

**Mechanical Equipment (A2)**

<table>
<thead>
<tr>
<th>Lot of</th>
<th>Few</th>
<th>3</th>
</tr>
</thead>
</table>

**Electrical Equipment (A3)**

<table>
<thead>
<tr>
<th>Lot of</th>
<th>Few</th>
<th>3</th>
</tr>
</thead>
</table>

**Air-tight (G)**

| Yes | No | 2 |

**Ignition of Combustible Material (B, C2); How Inflammable?**

<table>
<thead>
<tr>
<th>Very: $&lt; 20^\circ$C</th>
<th>Normal (oil, $&lt; 250^\circ$C)</th>
<th>Difficult (paper, electr. equipm.)</th>
<th>Very difficult (wood etc.)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>2</td>
<td>3</td>
<td>4</td>
</tr>
</tbody>
</table>

**Distribution of Combustible Material in the Room (F)**

<table>
<thead>
<tr>
<th>Everywhere</th>
<th>Mostly</th>
<th>1/2</th>
<th>Few</th>
<th>None</th>
<th>5</th>
</tr>
</thead>
</table>

**Is the Fire Load Sufficient to Destroy the Components in the Room? (D)**

<table>
<thead>
<tr>
<th>Yes</th>
<th>No</th>
<th>2</th>
</tr>
</thead>
</table>

**Automatic Sprinkler**

**Other Extinguishing Equipment**

**Fire Alarm**

**Notes (Quality of Fire Load etc.)**
VALUES FOR THE PARAMETERS OF BERRY'S MODEL

<table>
<thead>
<tr>
<th>Code and explanation for the parameters</th>
<th>Values for the parameters</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1</td>
</tr>
<tr>
<td><strong>A</strong> Presence of ignition sources</td>
<td></td>
</tr>
<tr>
<td>A1 Persons</td>
<td>0.70</td>
</tr>
<tr>
<td>A2 Mechanical equipment</td>
<td>0.50</td>
</tr>
<tr>
<td>A3 Electrical equipment</td>
<td>0.30</td>
</tr>
<tr>
<td><strong>B</strong> Ignition</td>
<td></td>
</tr>
<tr>
<td><strong>C</strong> Detection and self-extinguishing of fire</td>
<td></td>
</tr>
<tr>
<td>C1 Persons</td>
<td>0.99</td>
</tr>
<tr>
<td>C2 Self-extinguishing</td>
<td>0.50</td>
</tr>
<tr>
<td><strong>D</strong> Is the fire load sufficient to destroy the safety related component?</td>
<td></td>
</tr>
<tr>
<td><strong>F</strong> Distribution of the fire load</td>
<td></td>
</tr>
<tr>
<td><strong>G</strong> Air-tightness of the room</td>
<td></td>
</tr>
</tbody>
</table>

The relative combustion probability, $P_K$, of the room $k$ is

$$P_K = A \cdot B \cdot (1-C_1 \cdot C_2) \cdot D \cdot (1-F \cdot G),$$

where

$$A = 1-(1-A_1) \cdot (1-A_2) \cdot (1-A_3)$$
Core damage frequency due to fires 3.7E-6

- Cable rooms (2/4 trains) and Inadv. depressurization 7.3E-7
- Cable rooms (2/4 trains) 6.6E-7
- Pump room, left aux building 3.7E-7
- Pump room, right aux building 3.5E-7
- Cable rooms (1/4 train) 3.1E-7
- LOOP 2.7E-8
- Others 1.2E-6
### EXAMPLES OF FIRE PROTECTION ACTIONS AT THE NPP TVO I/II

<table>
<thead>
<tr>
<th>Room</th>
<th>Room code</th>
<th>Fire initiator class</th>
<th>p₂ old</th>
<th>p₂ ne</th>
<th>Spr</th>
<th>Coa</th>
<th>Wal</th>
<th>Note</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cable shaft BD</td>
<td>E2.44</td>
<td>3;12;24</td>
<td>0,1</td>
<td>0,1</td>
<td>X</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cable shaft AC</td>
<td>E92.43</td>
<td>3;12;24</td>
<td>0.9</td>
<td>0.1</td>
<td>X</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cable room BD</td>
<td>E2.28 a)</td>
<td>3;12</td>
<td>1,0</td>
<td>0.1</td>
<td>X</td>
<td>X</td>
<td></td>
<td>E2.28 has been divided to two separate fire departments</td>
</tr>
<tr>
<td>Cable room D</td>
<td>E2.28 b)</td>
<td>12</td>
<td>1,0</td>
<td>-</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cable room AC</td>
<td>E2.27 a)</td>
<td>3;12</td>
<td>1,0</td>
<td>0.1</td>
<td>X</td>
<td>X</td>
<td>E2.27 has been divided to two separate fire departments</td>
<td></td>
</tr>
<tr>
<td>Cable room C</td>
<td>E2.27 b)</td>
<td>12</td>
<td>1,0</td>
<td>-</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pump room</td>
<td>H1.53/55</td>
<td>9</td>
<td>1,0</td>
<td>0.1</td>
<td></td>
<td></td>
<td>X</td>
<td>The power cables of the pumps have been partly protected with the fire retardant cable coating</td>
</tr>
<tr>
<td>Pump room</td>
<td>H1.54/56</td>
<td>10</td>
<td>1,0</td>
<td>0.1</td>
<td></td>
<td></td>
<td>X</td>
<td>&quot;</td>
</tr>
<tr>
<td>Cable room BD</td>
<td>X91.06</td>
<td>16;17; 18;19</td>
<td>0.9</td>
<td>0.1</td>
<td></td>
<td></td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Cable room AC</td>
<td>X91.07</td>
<td>16;17; 18;19</td>
<td>0.9</td>
<td>0.1</td>
<td></td>
<td></td>
<td>X</td>
<td></td>
</tr>
</tbody>
</table>

- **p₂, old** The value of the conditional probability p₂ (the propagation of the fire from one train to another) before the fire protection actions
- **p₂, new** The value of the conditional probability p₂ after the fire protection actions
- **Spr** Changes in the sprinkler systems of the room (more sprinkler heads, quick response sprinklers, changing the location of some heads, installing sprinklers to new areas)
- **Coa** Protection of the cables with the fire retardant cable coating
- **Wal** Division of the room to a number of separate fire departments
AN APPROACH FOR THE APPLICATION OF PROBABILISTIC FIRE RISK ANALYSIS IN GERMAN NUCLEAR POWER PLANTS

H.P. Berg
Bundesamt für Strahlenschutz, Salzgitter
G. Breiling, H.H. Hoffmann
ABB Reaktor GmbH, Mannheim
Federal Republic of Germany

Abstract

The evaluation of the risk associated with fires in nuclear power plants has attracted considerable attention in the last several years. While it was expected earlier that this risk may be very small and, therefore, dominated by other risks, more detailed safety assessments - in particular in the USA after the famous Browns Ferry fire incident - have shown that the consequences from fires could be significant. Methodologies considering fire in probabilistic safety analyses have been evolving.

Such analyses allow the identification of weak points based on quantitative results and thus will provide a strictly safety-related basis for setting priorities of improvement measures.

Hence, a probabilistic fire risk analysis can act as an additional tool for the safety evaluation of a nuclear power plant.

In order to provide a basis for further discussions on benefits and limits of a probabilistic fire risk analysis (PFA) in Germany, current methods are investigated resulting in an approach for a possible procedure of PFA in German nuclear power plants including a first draft of a respective guide.

The proposed complete probabilistic fire risk analysis has to be performed in different steps. Major tasks are the qualitative screening process to identify critical fire zones and the quantitative event tree analysis in which the fire caused frequency of initiating events and different core damage states will be determined.

The proposed method fits into the methodical frame of a probabilistic safety assessment (PSA) provided by the German PSA guideline.

The combination of the probabilistic fire risk analysis with the level 1 PSA of the respective nuclear power plant will allow to calculate the fire induced contribution to the core melt frequency and can provide sufficient answers with respect to plant vulnerability.
1. Introduction

Fires are rather localized phenomena in a nuclear power plant; i.e., their direct damage is almost always confined to a few locations within the plant.

On the other hand, the main concern of internal fires is their potential to simultaneously damage electrical components, especially control and power cables, of redundant safety equipment; fire effects have therefore a considerable influence on plant safety. The well-known cable spreading room fire at Browns Ferry Station (USA) in 1975 demonstrated that fire can be a potential accident initiator.

A modern assessment of the effects of fire has to consider all sources of combustible material and the quantity present, the potential for ignition and the adequacy of protection, both passive (barriers) and active (firefighting) to allow the frequency and extent of a fire to be determined. Such assessment have to take into account the results of investigations showing that most of the fires are caused by ignition of oil, H₂ leakages or faults in electrical equipment and cables.

Thus, unlike the internal event risk analysis, a fire risk analysis will have to consider almost all of the plant locations that contain safety-related equipment but also locations that contain the power and control cables of the safety-related equipment.

Results of fire risk assessments, in particular in the US, have shown that fires must be considered as a significant core damage contributor, with contribution to overall core damage frequency ranging anywhere from insignificant to 55% of the total frequency. It is found that although functional unavailability of redundant trains is relatively small, internal fires can reduce the level of redundancy by damaging one or two of the redundant systems, or their supporting systems.

As a consequence of the results of the performed probabilistic fire risk assessments, in USA and partly in Europe, i.e. Finland, Spain, Sweden and Switzerland, a full scale PSA including external events is common practice.

The safety criteria for nuclear power plants of 1977 in the Federal Republic of Germany [1] include safety-related design principles. Criterion 1.1 "Principles of Safety Precautions" requires, among others, reliable safety systems. In a footnote to this principle it is stated that the reliability of these systems and components shall be determined with the aid of probabilistic methods as far as the required accuracy can be achieved according to the state of science and technology.

In the meantime, probabilistic safety assessment (PSA) has become a very powerful tool to investigate, verify and improve safety practices. Therefore, the German regulatory authorities are increasingly making use of PSA in safety evaluations of nuclear power plants. Moreover, a level 1+ PSA is required as one part of the planned periodic safety reviews of all nuclear power plants in operation about every ten years.
Since an important goal is to have PSAs that are comparable as far as possible a common source of guidance for performing a PSA is desirable in order to obtain results with a comparable quality level.

Therefore, the committee "Probabilistic Safety Analysis (PSA) for NPP's" has prepared a first version of a guideline for performing such a PSA which has been published in 1990 [2]. In that guideline an analysis of plant internal fires and external events is explicitly not required referring to the unavailability of appropriate models and methods at that time. A revision of this guideline is underway as part of the guidelines for periodic safety assessments of nuclear power plants; the revised PSA guideline is mainly based on [2] and shall not contain statements on probabilistic fire risk analysis at that stage.

The Gesellschaft für Reaktorsicherheit published detailed work on fire risk quantification that serve as basis for the fire analyses done for the German Fire Risk Study [3]. Within the scope of the German Risk Study, Phase B [4] a probabilistic fire risk analysis (PFA) has been performed for a typical 1300 MWe PWR plant. Moreover, the recently completed first phase of a comprehensive safety analysis for a typical 1300 MWe BWR plant of 72-type also contains a first evaluation of fire risks, partially based on generic US data [5].

2. International experiences in probabilistic fire risk analysis

In the USA fire risk analysis methodology for nuclear power plants has been evolving and improved for over a decade, and a dozen NPPs have applied them during the last ten years. Many of these studies show that fire risk constitutes a significant percentage of external events as well as the total core damage frequency.

Past PSAs show that the risk contribution of fires to core damage is very plant specific because of the different physical layouts of different plants such as physical separation, redundancy of the safety and nonsafety systems, and their degree of compartmentation. The risk contribution of fires to core melt of the "N + 1" Seabrook Station (with two independent redundant safety systems) is as high as 2.5 x 10⁻⁵ per year (11% of the total core melt frequency). For the South Texas Project, a typical three-train plant ("n+2" redundancy concept), a mean core melt frequency for all fire initiators of < 2.7 x 10⁻⁶ per year has been calculated (1.4% of the total core melt frequency). For a German "N+3" plant, the risk contribution of fire events is in the order of 10⁻⁷ per year (= 14% of the total core melt frequency) [6].

One of the recent methodologies combines engineering judgment, statistical evidence, fire phenomenology, and plant system analysis. The intended use of this analysis is for estimating the probability of a catastrophic effect on plant safety due to fire.

The latest probabilistic approach is the Fire Induced Vulnerability Evaluation (FIVE) Methodology by the Electric Power Research Institute (EPRI) [7].

This methodology applies qualitative and quantitative screening techniques to estimate the frequency of dominant fire initiators
and convolve these with the system model of the level 1 PSA of the plant to come up with the core damage frequency due to fire. This methodology uses deterministic design information, combining it with an existing probabilistic analytical system model of the plant and structured look-up tables. It utilizes the benefits of both qualitative and quantitative analyses.

As a direct outgrowth of the comprehensive US fire research activities which included significant involvement in probabilistic risk assessments for fire in a nuclear power plant a fire risk scoping study has been performed [8]. The specific objectives of this study are

- to review and requantify fire risk scenarios from four probabilistic fire risk assessments in light of updated data bases and updated computer fire modeling capabilities,

- to identify potentially significant fire risk issues that have not been previously addressed in a fire risk context and to quantify the potential impact of those identified fire risk issues where possible,

- to review current fire regulations and plant implementation practices for relevance to the identified unaddressed fire risk issues.

In the European countries such comprehensive experience in probabilistic fire risk analysis doesn't exist. Nevertheless, there are attempts, for example, in Finland, Sweden and Switzerland.

Within the ABB group fire risk analysis work was done for BWR, PWR and radio-chemical facilities. Early work was sponsored by the Swedish Nuclear Power Inspectorate [9, 10]. Today's activities concentrate on backfitting programs of older plants and analyses of advanced reactor designs (System 80+, BWR90)

3. Proposal for a German approach to probabilistic fire risk analysis

In order to provide a basis for further discussions on benefits and limits of a probabilistic fire risk analysis (PFA) in Germany, current methods are investigated resulting in an approach for a possible procedure of PFA in German nuclear power plants including a first draft of a respective guide.

The models used in the analysis are, in spite of the simplifications and uncertainties, accurate enough to identify plant vulnerabilities due to fire incidents, and to assess plant improvements with respect to fire protection.

The proposed complete PFA has to be done in different procedural steps. Major tasks are the qualitative screening process to identify critical fire zones and the quantitative analysis which are described below in more detail.
3.1 Screening analysis

The qualitative screening process to identify critical fire zones is an important first step within a PFA. On the one hand, the screening analysis should not be so conservative that an unmanageable number of fire scenarios remain for the detailed quantitative analysis. On the other hand it must be ensured that all relevant areas are investigated within the quantitative analysis.

In the first step of the screening analysis, detailed data must be collected in all rooms of the plant to identify the essential and the critical fire zones.

A fire area is defined as a building or a part of a building which is sufficiently bounded by fire barriers preventing fire propagation to adjacent buildings or parts of buildings [11].

A fire zone is a subdivision of a fire area preventing propagation or unallowed consequences to other subdivisions.

An essential fire zone contains:
- equipment related to power operation,
- safety related equipment,
- fixed or transient combustibles.

A critical fire zone is an essential one, in which a fire
- damages at least one safety-related component or system and
- causes a safety-related initiating event.

Data have to be collected for about 1000 rooms; during the screening process it can be expected to reduce the number of rooms to around 10 - 30 which may deliver a remarkable contribution to the fire initiated core melt frequency.

The data collection comprises:
- Fire protection maps in which all buildings are subdivided in numbered fire zones and rooms. The definition of a "room" is not necessarily related to fire rated enclosures but takes into consideration distance, system installation areas and range of suppression systems.

- Inventory lists of fixed and transient combustible loadings. The fire load per floor area [kWh/m²] will be calculated for each combustible material such as oil, cable insulation, tissues, clothes, paper, filters, plastic etc. The insulation of cables within metal conduits or cabinets or electrical motors are to be considered, too. If the fire load is less than 7 kWh/m² [12] the room will be neglected.

- Lists of electrical and sensing equipment per room. These lists contain all electrical consumers, such as valves, pumps, drives, motors, cranes, electrically actuated dampers, heaters, ventilators, temperature, pressure and level measurement equipment etc. In most plants these data are available on computer files.
Tables of cable routings for safety and operational systems. These tables contain the location of the components (room number), the electrical bus providing power to the components, rooms crossed by cables (room numbers), redundancy assignment etc. The assumption is, that the failure of the cable in any room can lead to the same consequence as the failure of the component itself. Both, power and signal cables are to be taken into account.

- A list of initiating events, mitigating systems and functions of the existing level 1 PSA.

The large amount of necessary data requires the use of a PC based software package in order to handle the data in an appropriate way.

In case the above information does not exist, the experiences from US nuclear power plants show that numerous plant walk downs are necessary for such a data collection.

After the data collection including allocation of combustible loading, allocation of safety and operational components and systems to rooms and fire zones is completed, the essential and critical fire zones (rooms and pairs of rooms) must be determined.

This part of the screening process starts with the identification of all rooms for which at least one of the following 3 criteria is fulfilled:

- Fire load > 25 kWh/m²,
- Room contains safety related equipment or cables of such equipment,
- Room contains operational or sensing equipment of the reactor protection system or cables of such equipment.

Rooms, for which all 3 criteria are fulfilled simultaneously, will be identified as essential fire zones/rooms. For rooms in which 2 out of 3 criteria are fulfilled, adjacent rooms are checked to identify pairs of rooms that fulfil all 3 criteria. The critical fire zones/rooms and pairs of rooms are selected based on the above described criteria.

Having screened out all unimportant rooms, a detailed and comprehensive data collection concerning fire protection will be done for the remaining critical ones as described in Table 1.

The data on room/zone geometry, ventilation conditions and combustible loading described in Table 1 are necessary informations because calculations have shown that the ventilation of the fire room/zone plays an important role for the fire propagation in the respective area.
Table 1: Additional data for critical fire zones/rooms

<table>
<thead>
<tr>
<th>Data on building structures</th>
</tr>
</thead>
<tbody>
<tr>
<td>• fire resistant enclosure</td>
</tr>
<tr>
<td>• rated barriers such as doors, sealed openings, heat and smoke removing systems</td>
</tr>
</tbody>
</table>

Data on installations

<table>
<thead>
<tr>
<th>Data on installations</th>
</tr>
</thead>
<tbody>
<tr>
<td>• equipment for fire detection</td>
</tr>
<tr>
<td>• equipment for fire alarm</td>
</tr>
<tr>
<td>• equipment for fire suppression</td>
</tr>
</tbody>
</table>

Data on administration

<table>
<thead>
<tr>
<th>Data on administration</th>
</tr>
</thead>
<tbody>
<tr>
<td>• trained personnel</td>
</tr>
<tr>
<td>• fire brigade</td>
</tr>
<tr>
<td>• fire protection inspection procedures</td>
</tr>
<tr>
<td>• fire protection maintenance procedures</td>
</tr>
</tbody>
</table>

Data on room/zone geometry

<table>
<thead>
<tr>
<th>Data on room/zone geometry</th>
</tr>
</thead>
<tbody>
<tr>
<td>• floor area</td>
</tr>
<tr>
<td>• surface</td>
</tr>
<tr>
<td>• content</td>
</tr>
</tbody>
</table>

Data on ventilation conditions, e.g.

<table>
<thead>
<tr>
<th>Data on ventilation conditions, e.g.</th>
</tr>
</thead>
<tbody>
<tr>
<td>• natural or forced convection</td>
</tr>
<tr>
<td>• openings</td>
</tr>
<tr>
<td>• fire dampers</td>
</tr>
</tbody>
</table>

Data on combustible loading, e.g.

<table>
<thead>
<tr>
<th>Data on combustible loading, e.g.</th>
</tr>
</thead>
<tbody>
<tr>
<td>• quantity</td>
</tr>
<tr>
<td>• location</td>
</tr>
<tr>
<td>• physical properties</td>
</tr>
<tr>
<td>• burning rate</td>
</tr>
</tbody>
</table>
3.2 Quantitative analysis

For each critical fire zone or room an event tree will be developed with nodes for fire initiation, ventilation of the room, fire detection, fire suppression and propagation. The development of a fire, once initiated, depends on the availability of the fire detection and suppression systems. The availability of the fire barriers influences the fire propagation to adjacent rooms. Fig. 1 shows a typical event tree for an oil fire in a diesel generator room.

![Event Tree Diagram]

Fig. 1: Example of an event tree for an oil fire in a diesel generator room

For the fire ignition frequency the latest published data base [7] for US plants will be used, since it includes 800 fire incidents caused by fixed and transient ignition sources due to normal plant operation and maintenance activities from 114 BWR and PWR units representing a total sample size of 1270 reactor-years. The older data base, covering a period from 1968 to 1978, gives an average fire frequency per year and plant of 0.17 [13] (only such fires are counted, which are relevant to the plant safety). The new data base, covering a period up to 1988, gives a number of 0.38, counting fires in all plant locations during power operation and maintenance activities.

The German data base covers only 55 fire events in 18 plants (including test and demonstration plants) from their first criticality until November 1983. The average fire frequency per plant and year of 0.40 [14] compares very well with the US figure.
In principle the US data base can be applied to German plants, however certain adjustments are recommended to allow for design differences. One such difference is that in German plants all switchgear is concentrated in one switchgear building, and no motor control centers are placed with mechanical equipment like pumps etc.

To calculate the fire frequency for a single room in a building additional weighting factors based on the fractional amount of ignition sources, the weight of cable insulation, the number of relevant fire zones and special factors for the ignition sources [7] are necessary.

The nodes "early and late fire detection", "early and late fire suppression", "room closure" are quantified according to [3, 15]. The node "equipment damage caused by fire suppression agent" is introduced due to a NRC request [16] and a 0.3 probability for suppression damage is assumed in case agent-susceptible equipment is in the room.

As a result of the PFA five different damage states are distinguished:

- Minimal damage (a) occurs when early detection and early suppression are successful. Only the specific component, cable or combustible, where the fire started, is damaged.

- Partial damage (b) exists when early detection is successful, the use of portable extinguishers fails and the use of a hose or deluge system is successful to suppress the fire. Safety-related components close to the point of fire ignition are assumed to be damaged.

- Partial damage (c) corresponds to partial damage (b) but the suppression agent, mostly water, has damaged the agent susceptible equipment in the room.

- Total damage (d) exists, when
  - early detection is successful and early suppression fails, but late suppression with fixed installed equipments prevents propagation to a neighbouring room or fire zone,
  - early/late fire detection fails/is successful and late fire suppression is successful.

All safety related equipment and cables of critical components in the room are damaged.

- Total damage and propagation (e) happens when all fire protection measures fail to prevent the propagation to adjacent rooms. All safety-related equipment is damaged in the neighbouring rooms.

For each critical fire zone/room the following results are obtained:

- Frequency and nature of fire initiated transients.
A list of damaged equipment, categorized corresponding to the damage states (a), (b), (c), (d), and (e).

Frequency of the damage states.

This will give an insight in the fire vulnerability of the plant. But finally the combination of the PFA with the level 1 PSA will allow to calculate the fire induced contribution to the core melt frequency. For each initiating event the fire caused frequency will be summed up and specified as input to the corresponding event tree of the level 1 PSA. Additionally the damage state of the safety related equipment has to be introduced into the fault trees. For all components that are not affected the statistical failure behaviour is used. The core melt frequency is calculated for each transient as the sum of the single event core melt frequencies. The total core melt frequency is obtained by summing the contribution of all transients.

4. Further procedure

The investigations on currently applied models and methods in probabilistic fire risk analysis have resulted in a proposal for a possible approach to apply also probabilistic methods in the case of evaluating the consequences of fire events. The possible procedure is described in a first draft of a guideline for such a probabilistic fire risk assessment.

The next step is the discussion of this proposal with other German PSA experts as it has been the case in preparing the first version of the PSA guideline. Afterwards, the PSA of fire events has to be discussed with the Federal States in the respective committees which has to be initiated by the Federal Ministry of Environment, Nature Conservation and Nuclear Safety supported by the Bfs. Also the Reactor Safety Commission will be invited to prepare their comments and recommendations.

If the proposed (or maybe slightly changed) procedure is approved by the different expert groups, it is intended to implement such a procedure as part of the level 1 PSA and to use the results of PFA within the periodic safety review as additional informations to the deterministic analysis. Then, the probabilistic results can be taken into account in the evaluation of divergences concerning fire events found during the periodic safety review.

5. Conclusions

Based on the experience gained in the USA since 1968 the frequency of reportable fires during plant operation is approximately 0.38 per reactor-year. German investigations have led to comparable results (0.4 per reactor-year). Hence, a plant can expect 15 fires during its design live of about 40 years. Up to now, no fire has led to core damage, but several incidents represent significant precursors.

Some of the published PSAs have included an analysis of fires and shown that fires contribute up to 55 % to the overall core damage frequency [17]. Plant modifications in Indian Point 2, Limerick and Haddam Neck reduced the fire induced core damage frequency up
to a factor of 10. But the risk from fire is still considered as a contributor to the overall risk.

The analysis for Haddam Neck for instance shows that the core melt frequencies are highest for the auxiliary building and the switchgear room [18] and lowest for the control room and the diesel generator building.

For older German plants the licensing authority asks for an updating of the fire protection status according to the current state of science and technology. The actual status will be compared to the required status outlined in the applicable rules and regulations. Such analyses are under development. Resulting deficiencies are catalogued, evaluated and discussed with the utility and with the competent authority. In this context a PFA can help to evaluate the divergences and, may be, existing safety deficiencies, to set priorities for fire protection improvement measures, as well as to determine the urgency of remedial measures.

References


COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS
PRINCIPAL WORKING GROUP N° 1 SPECIALIST MEETING
ON
"FIRES AND FIRE PROTECTION SYSTEMS
IN NUCLEAR POWER PLANTS"

Cologne, Germany
6th–9th, December, 1993

USE OF DETERMINISTIC AND PROBABILISTIC
TECHNIQUES FOR DECISION MAKING PROCESS:
COMPLIANCE WITH APPENDIX R TO CFR50
ABOUT FIRE PROTECTION FEATURES

Mª Dolores Morales Dorado

Francisco Javier Sanz Román

C.N. ALMARAZ
CLAUDIO COELLO, 123
28006 MADRID (SPAIN)
Phone: (1) 4314222
Fax: (1) 4357310
1. INTRODUCTION

The design, construction, operation and maintenance of a nuclear power plant is accomplished through a number of programs, procedures and administrative controls, most of them subjected to regulatory requirements, focused on ensuring adequate protection of public health and safety.

During the licensing process of the existing actual plants, deterministic criteria were used to define the bounding of the transients and accidents that the plant has to be able to afford maintaining the level of safety. This method gives a reasonable guarantee that the design can accommodate the uncertainties associated to the analysis.

Additionally, and when the plants are in commercial operation, new regulatory requirements appear, not considered in the initial design, that is necessary to take into account in a later stage, with the inherent difficulties that may appear.

During the last years, as a consequence of regulatory requirements in many countries, including USA and Spain, the nuclear power plants have developed their specific probabilistic safety analysis (PSA), that are a complementary tool to the deterministic existing analysis.

PSA include a very detailed and systematic analysis of all the possible initiating events (both transients and accidents), that can occur in the plant, together with the response of the mitigation systems to those incidents, taking into account possible human errors, components failures, unavailability due to test and/or maintenance, and dependencies and common cause failures, everything analyzed in a specific way for each plant. This allows to consider this type of studies as a backup tool in the decision making process because they allow to evaluate the impact of any change in design or procedures at the level of component, system, accident sequence, core damage and in some cases, releases to the environment.

In the case of Almaraz NPP, and based on its Probabilistic Safety Assessment approved by the Spanish Nuclear Regulatory Body (Consejo de Seguridad Nuclear, CSN) in April 1991, several applications have been done as a backup to deterministic safety evaluations, being one
of the most important cases the analysis of compliance with Appendix R to 10CFR50 about fire protection features, based on the deep knowledge of cables routings associated to safety equipments through the different areas and buildings of the plant, and the potential impact that a fire on them would have in the plant safety.

According to that, and taking into account both the probabilistic fire analysis performed (including the identification of the transients that can occur in each fire zone, as well as the possible fire origins and transient fuels) and the deterministic fire analysis available (including the calculation of the thermal load and the estimation of the duration of a generalized fire in each zone), the associated risk in each case has been evaluated, and therefore the impact of that risk in the fire protection proposed, in order to guarantee the fulfillment of the safety functions necessary for safe shutdown of the plant.

2. GENERAL REQUIREMENTS

Appendix R to 10CFR50 indicates that fire protection features have to be installed to guarantee safe shutdown capability for the plant.

These features will be able to limit fire damage so that one train of systems necessary to achieve and maintain safe shutdown conditions (not standby in the case of Almaraz NPP) from either the control room or emergency control station is free of fire damage, and systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station can be repaired within 72 hours.

In the case that redundant trains of systems mentioned above coincide in the same fire area, there are different acceptable means of ensuring that one of those redundant trains is free of fire damage:

- Separation of redundant trains by a fire barrier having a 3-hour rating.
- Separation of redundant trains by a horizontal distance of more than 20 feet with no intervening combustible or fire hazards, and installation of fire detectors and an automatic fire suppression system in the fire area.
Separation of redundant trains by a fire barrier having a 1-hour rating, and installation of fire detectors and an automatic fire suppression system in the fire area.

From the above it is easy to understand that, for plants designed before the issue of this type of regulation, the possible fire protection features to install would cover most of the safety systems, taking into account that it was not a design criteria at that moment to separate redundant trains a distance greater to 20 feet.

Based on that, the strict application of the new rule would imply the massive installation of fire protection features over the plant, with a consequent increase of the degree of complexity and difficulty of the different maintenance tasks.

According to this, and taking into account that during the development of the probabilistic fire risk analysis, cable routings and equipment location had been analyzed in detail for all the safety systems, as well as the list of initiating events (transients and accidents) induced by fires in any fire area in the plant, Almaraz NPP decided to combine deterministic and probabilistic techniques to try to minimize the amount of fire passive protection features to install on cables trays and conduits, guaranteeing the fulfillment of basic safety functions necessary for achieving and maintaining safe shutdown.

In the next sections, the different steps of the reanalysis performed will be described, starting from the listings of significant fire areas or zones from the deterministic and probabilistic point of view.

3. BASIC ASSUMPTIONS

The main differences about basic assumptions related to the deterministic and the probabilistic fire risk analysis can be summarized in the following way:
a) Deterministic analysis (Appendix R to CFR50)

- The train with fire protection features does not fail randomly due to causes different from fire.

- The fire does not propagate between different fire areas if they have fire barriers installed.

- Loss of offsite power is postulated simultaneously with the fire.

b) Probabilistic analysis (PSA)

- The train with fire protection features can fail randomly due to other causes, with a probability of failure that is considered to evaluate the risk associated to the occurrence and effects of fire.

- The fire can propagate between different fire areas, assigning a probability of failure to the different existing fire barriers.

- Loss of offsite power is not postulated to occur simultaneously with the fire with probability one, but with the corresponding probability of random failure.

From the above it can be seen that both types of fire analysis, deterministic and probabilistic, are complimentary related to the consideration of the importance and extension of the potential damage caused by the fire, obtaining from the joint study a more complete, detailed and specific perspective of the different situations that can occur in the facility.

4. STAGES OF THE ANALYSIS

Taking into account the above considerations, the analysis of the plant, from the point of view of fire risk, has been structured in three stages:

a) Analysis of important fire zones from both deterministic and probabilistic point of view, that is, those zones included in the scope of the Appendix R because they contain equip-
ments associated to both electrical trains, and important in the results of the probabilistic fire analysis because of their high contribution to the core damage frequency due to fires.

b) Analysis of fire zones involved in the deterministic study to accomplish with Appendix R, but not significant from the probabilistic point of view because of their negligible contribution to the core damage frequency due to fires.

c) Analysis of fire zones not considered in the deterministic study for fulfillment with the Appendix R because they do not need redundant trains separation, but which appear to be important according to the results of the probabilistic analysis of fires.

According to the above considerations, each fire zone in the plant has been analyzed, developing specific proposals of fire protection for every zone, consisting in some cases on the installation of new protection features and in other cases on the verification that the existing protection features are enough to guarantee the safety of the plant. Additionally, in some zones improvements have been proposed to the existing detection and/or suppression systems, as well as the development of new procedures or modifications to the existing ones to handle local actuations and recovery of equipment, in the way to minimize the risk of core damage due to fires and to accomplish with the deterministic rules.

As an example, it can be mentioned that, included in the first group of zones, i.e., zones important from both the deterministic and the probabilistic point of view there are the control room, cable room, remote shutdown panels zone, electrical penetration room, essential service water intake, and the train A 6,3 KV switchgear room.

Included in the areas affected by Appendix R, but not important from the probabilistic point of view, are most of the auxiliary building areas and several areas of the electrical building.

The fire zones significant from the probabilistic fire risk analysis but not included in Appendix R are the train A 125 V.c.c. and 118 V.c.a.r. distribution room, the train B 6,3 KV switchgear room, the train A diesel-generator room and the turbine building.

As it has been described before, the reanalysis performed for each fire zone has led to a specific proposal of fire protection features for the zone, having documented the report associated in the following way:
1. Zone description

2. Justification of the proposal of fire protection for the zone

3. Summary of the actions to be performed

4. Considerations from the point of view of PSA

In the section of description of the zone it is included a brief description of it, location (building and elevation), principal cables associated to safety functions in the zone, existing detection and suppression fire systems, as well as an estimation of the thermal load associated to the zone.

In the section of justification of the proposal of fire protection for the zone, the different situations that can happen as a consequence of a fire in the zone are analyzed, as well as the basic safety functions that have to be separated or protected in each case, including the considerations made related to the different existing alternatives, based on an exhaustive analysis of all the cables present in the zone, the possible transient and/or installed fuels and the possible actions to be taken to recover equipments in the acceptable period of time considered in the rule.

In the section of summary of the actions to be taken, it is clearly indicated the passive protection features to be installed or the improvements in the fire protections systems presently existing if it is considered necessary, identifying specifically the cable trays and conduits affected, their dimensions (width or diameter and length) and the resulting protection surface.

In the section of considerations from the probabilistic point of view, the proposals performed are evaluated, indicating the change in the core damage frequency due to fires in the zone and in the overall core damage frequency, estimating in this way the reduction in the risk associated to the implementation of the proposed improvements.

Additionally, in the report for each zone, drawings associated indicating graphically the cable trays and conduits protected are included.
This set of fire protection proposals for all the zones in the plant has been presented to and evaluated by the Spanish Nuclear Regulatory Body (CSN), together with an implementation program, being now in the situation of execution of the corresponding design changes, after having got the final approval of the regulatory body, once incorporated to the analysis the comments performed.

In summary, 53 attachments to the global design modification related to accomplishment with Appendix R have been elaborated, 31 of them including the installation of passive protection features. Additionally, new procedures of recovery of equipment have been developed, as well as modifications to existing procedures to verify the adequate position of some components after fire in their cables.

5. DOCUMENTATION DERIVED FROM THE ANALYSIS

Due to the high volume of information that is necessary to handle when this type of studies are performed, it has been considered that it can be useful, for both the Almaraz engineering and operation personnel, to have available backup documentation including, for each fire zone, the systematic identification of its principal characteristics, location, principal equipment that it contains, adjacent zones and communication between them, expected severity of the fire, and possible damage to the cables associated to the equipments that perform the necessary safety functions to achieve safe shutdown of the plant, both power and control cables.

To get this objective, at this moment complete tables are being developed, including in a systematic and ordered way, the above information for each fire zone.

With the aim that this information will be useful for the different sections of Almaraz NPP staff, both from the central offices and the site, the final report will be structured in a way that the level of detail of the consultation could be gradual, starting in a general level of basic documentation about existing equipments in the zone, adjacent fire zones, potential transients caused by fires in the zone and potential damage to front-line systems, to go after that to more detailed levels identifying specifically the affected component in each case for each system.
and safety function and the cause of the damage: damage in its own power or control cables, or in its own cooling system, or damage in its support systems (power, control or cooling), leading in each case to different recovery strategies after a fire.

Additionally, the potential impact of the fire in the adjacent zones on the defined safety functions is analyzed and, depending on the characteristics of the communication between the zones, if those effects are of high or low probability.

It is expected that this information will be useful for training of plant personnel in the activities of fire protection.

6. SUMMARY AND CONCLUSIONS

The results of the analyses performed by Almaraz NPP lead to a set of specific proposals of fire protection features for each fire zone in the plant, based on the actual risk associated to each zone. These proposals have been presented to the Spanish Regulatory Body (CSN) and evaluated and approved by this organization, having incorporated in the study the comments performed during this evaluation.

At present, Almaraz NPP is implementing the protection features and improvements mentioned, having developed the procedures identified to handle potential local actuations and recoveries of equipments, with the aim to minimize the risk of core damage due to fires in the plant and at the same time to accomplish with the existing rules in the area of fire protection.
METHODOLOGY OF THE FIRE PROBABILISTIC SAFETY ASSESSMENT OF 900 MWe PRESSURIZED WATER REACTORS

Penh LAMUTH and Rémy BERTRAND
Institut de Protection et de Sûreté Nucléaire
FONTENAY-AUX-ROSES

1 INTRODUCTION

In the context of fire protection in technical buildings of French nuclear power plants, with the exception of the turbine building, three principles were adopted: prevention, detection and fire fighting. Their implementation makes it possible, on the one hand to limit the risk of fire, on the other hand to take suitable measures to prevent fire spread which would lead to simultaneous unavailability of systems ensuring the same safety function.

A fire Probabilistic Safety Assessment, called fire PSA, is being carried out by the Institut de Protection et de Sûreté Nucléaire (IPSN) to aid the safety assessment of operating 900 MWe PWR plants.

This study which considers the internal aggression resulting from a fire occurring in various location is the continuation of the level 1 Probabilistic Safety Assessment, called PSA 900, already performed by the IPSN for the same standardized series of plants, taking into account internal events likely to occur in the installation (random failures of equipment or human errors). The overall probability of core meltdown obtained by PSA 900 is of the order of 5.E-5/reactor-year (refer to [1]). Besides, the fire PSAs conducted in the USA have shown that a fire could play a major role in the total probability of core meltdown. For instance, for the SURRY reactor, the calculated fire-induced core meltdown probability is about 1.1.E-5/reactor-year (refer to [2]).

The general objective of the fire PSA is to confirm and to supplement the deterministic analysis on which the reactor design and the fire protection provisions are based. It aims at a better appraisal and prioritization of nuclear risks related to fire and should make it possible in particular to highlight the strong points and the weak points of the installation and of its operation, in its various operating states, as regards the protective measures taken against fire hazards.

The fire-PSA will notably make it possible to:

- provide useful information for safety analyses such as, for example, the rooms in which the contribution of a fire is the most important for the probability of core meltdown,

- conduct the necessary analyses within the framework of safety reassessments.

It should contribute to the assessment of design options relative to fire protection for future reactors.

The reactor considered for the fire PSA is unit 1 of Le Blayais nuclear power plant.
2 METHODOLOGY OF THE FIRE-PSA

The objective is to quantify fire-induced core meltdown sequences and the corresponding probabilities.

It is noteworthy that fires which may result from external hazards such as earthquakes are not considered in the fire PSA.

The following main tasks (refer to Figure 1) must be performed.

2.1 INVENTORY OF DATA DERIVED FROM EXPERIENCE FEEDBACK

The objective is to draw lessons from experience feedback of fire events making it possible to develop statistics as regards:

- the occurrence frequency of a fire,
- the failure of the installed equipment, directly or indirectly subjected to a fire, and of the communicating elements between rooms,
- the failure of fire detection and fire fighting systems,
- the failure of protection devices provided to deal with common points of trains A and B,
- the time taken by the intervention team to extinguish a fire.

This work is based on French and foreign experience feedback.

Fire events which occurred during construction or in the preoperational mode of operation are not included in the inventory.

2.2 INVENTORY OF FUNCTIONAL DATA FOR EACH ROOM

The objective is, on the one hand to locate fire initiators and the safety-related equipment, on the other hand to collect the data necessary for the numerical simulation of the development of a fire in a room and its possible spread to adjacent rooms.

Experience feedback shows that a fire may break out in an ordinary room which does not contain safety-related equipment but that in case of spread to an adjacent room containing safety-related equipment, it may jeopardize safety.

The inventory must therefore be as exhaustive as possible and this in every room of technical buildings.

These buildings are divided into a number of zones including one or several rooms with a view to the following identifications.

2.2.1 IDENTIFICATION OF EQUIPMENT

All the equipment (mechanical, electrical, detectors, extinguishers), safety or not-safety-related, is listed and its functions are known.
2.2.2 IDENTIFICATION OF FUELS AND FIRE INITIATORS

The equipment which may be involved in combustion or be at the origin of a fire is identified according to the two following criteria:

- fuels:
  * solvents used during maintenance operations,
  * lubricating oils for pumps,
  * diesel engine oil,
  * materials used for cable sheaths, etc.

- fire initiators:
  * mechanical equipment (e.g. motors for pumps and fans),
  * electrical equipment (e.g. 6.6 kv switchboard),
  * welding stations, hot spots (e.g. pumps in operation), etc.

2.2.3 IDENTIFICATION OF COMMUNICATING ELEMENTS BETWEEN ROOMS

Such elements (doors, dampers, cable runs, ventilation ducts, etc.) are listed for fire spread studies. Their possible failure (non closing, ageing, etc.) is the subject of a fire spread study.

Finally, the drawings mentioning the dimensions of the rooms, the electrical connections and other elements necessary for the fire numerical simulation study such as ventilation rates, locations of equipment and positioning of cable routing are also identified.

2.3 OCCURRENCE FREQUENCY OF A FIRE

The fire initiators, known during the experience feedback analysis, are grouped into families. Moreover, the inventory of functional data makes it possible to identify the rooms including one or several fire initiators belonging to these families.

It is thus possible to assess, for an initiator in a given room, the occurrence frequency of a fire. This assessment is carried out for all the initiators and rooms in which they are to be found.

2.4 SELECTION OF CRITICAL ZONES

The number of zones which are the subject of the inventory of functional data is considerable. These zones are selected and only those where a fire is likely to develop and jeopardize safety are kept. The selected zones are called critical zones and their selection is made according to the following criteria:

- a critical zone contains safety-related equipment,
- the loss of equipment of a critical zone is at the origin of an initiator or of the total loss of a redundant system,

- a zone in which the fire contribution to the overall probability of core meltdown cannot be neglected.

Inside each critical zone, the equipment for which destruction would have repercussions on core meltdown probability has to be identified.

2.5 FIRE SCENARIOS

A fire scenario is a chain of subsequent events the first of which is the fire initiating event (starting point of the fire); the others are called "generic events" and generally involve the following possibilities:

- report of a fire by detectors on the spot or by human presence,

- fight by equipment on the spot or by intervention team,

- natural extinguishing by lack of fuel or of oxygen,

- fire spread due to lack of intervention and failure of communicating elements.

In case of fire spread to an adjacent room which may be located in another critical zone, it is necessary to extend the fire scenarios, taking account of the generic events in the new room.

The scenarios including these possibilities are developed in all the critical zones and are represented by event trees called fire event trees. An example of fire event trees in a room of a critical zone is shown in Figure 2.

2.6 STUDIES OF FIRE DEVELOPMENT

The objective is to predict, by fire numerical simulation in the critical zones, how long it will take to damage the equipment directly or indirectly subjected to the effect of a fire.

Considering the particular feature of the plant design, a specific numerical simulation programme, called FLAMMES programme, is developed by the IPSN to simulate the development of a fire in a room which has the following characteristics:

- parallelepiped form,

- equipped with several horizontal and vertical communicating elements which may result in fire spread to adjacent rooms,

- confined or ventilated mechanically or naturally,

- containing several fuel sources which may, on the one hand be at ground level or overhead, on the other hand burn not simultaneously,
- containing target equipment located in or outside the plume of the fire location.

The code also makes it possible to simulate fire spread to adjacent rooms.

The design of the FLAMMES programme is based on the assumption of a three-zone model:
- cold zone,
- hot zone,
- plume of the flame (above the flame and under ceiling).

Figure 3 shows the simulation possibility of the programme.

2.7 PROBABILISTIC QUANTIFICATION OF FIRE SCENARIOS

The objective is, on the one hand to quantify fire scenarios leading to the partial or total destruction of the targets of a room and to a possible spread to the adjacent room, and on the other hand to assess target destruction probabilities.

The fire scenarios were modelled by fire event trees. Quantifying these scenarios requires the assessment of the probability of generic events. Depending on the nature of the generic event, this assessment will be carried out either directly from experience feedback or by using the failure tree method.

Human errors made during operations specific to fire (detection confirmation, fire fight, etc.) are also the subject of an analysis and quantification. Their assessment is based as far as possible on the specific information derived from experience feedback and tests performed on a simulator.

Once each fire scenario is quantified, the damage probability of each target is assessed.

In addition to fire destruction, the possibilities of damage of the target by the implemented extinguishing means (e.g. spray of a relaying circuitry cabinet with water) must be considered.

The model developed for PSA 900 will be used, after adaptation, to quantify fire-induced core meltdown sequences.

2.8 QUANTIFICATION OF CORE MELTDOWN SEQUENCES

This final work consists in assessing the probability of core meltdown sequences likely to result from a fire. It includes requantifying:

- human factors associated with operation teams;
- fault trees, taking into account the fire damage probabilities of equipment located in critical zones;
- event trees to assess the probability of core meltdown.

3 DIFFICULTIES ENCOUNTERED

Three main difficulties were encountered.

3.1 NUMERICAL DATA OF CALCULATIONS

Collecting all data requires important bibliographic enquiries and representative tests.

The following main information should therefore be as precise as possible:

- statistical data derived from experience feedback,
- data relative to human factor,
- data relative to combustion (e.g. combustion rate, cable pool fires),
- data relative to environmental conditions (e.g. ventilation rate, open door),
- damage criteria (temperature or critical heat flux) of safety-related equipment subjected to a fire.

3.2 INTERPRETATION OF THE RESULTS OF FIRE NUMERICAL SIMULATION

The main parameter sought is the time taken to damage the safety-related equipment subjected to a fire during its development and extinguishing phases, for each selected scenario.

In view of the above-mentioned difficulties, the simulation results cannot be directly used without the advice of the fire specialist to make sure of the representativity of the physical phenomenon.

Parametric studies are required to determine in each critical zone the minimum period of time taken to damage the equipment.

3.3 BEHAVIOUR OF ACTUATORS AND OF THE POWER DISTRIBUTION EQUIPMENT

A fire may damage directly and indirectly the equipment.

The analysis of repercussions on actuators due to the damage of instrumentation and control cables requires therefore particular attention.

Furthermore, cables damaged by a fire may also have consequences on the power distribution equipment in case of short circuit or circuit breaking.

Consequently, a specific analysis is necessary.
4 CONCLUSION

Carrying out the fire-PSA using the above-mentioned methodology represents a sizeable amount of work. It requires the assistance of several specialists in:

- fire physics,
- PWR operation in incident and accident situations,
- statistics and probabilistic analysis,
- reactor operation,
- human factor.

Continuous contacts are established with Electricité de France which carried out the fire PSA for 1 300 MWe PWRs so as to harmonize approaches.
REFERENCES

Main report, April 1990.

October 1990.
SIMULATION POSSIBILITY OF THE FLAMMES PROGRAM

THREE-ZONE MODEL

THERMAL ZONE

Target

Hot Gas Layer

Gas Layer Convection

Plume Gases

Outflow of Hot Gases (Open Door)

Inflow of Ambient Air (Open Door)

Natural or Forced Ventilation

Flame Radiation

SEVERAL SEATS OF FUEL

FIRE SPREAD TO ADJACENT ROOM

FIRE

398

Figure 3
<table>
<thead>
<tr>
<th>Name</th>
<th>Organisation</th>
<th>Nationality</th>
<th>Phone/Fax</th>
</tr>
</thead>
<tbody>
<tr>
<td>Hans Olov (Olle) Andersson</td>
<td>Forsmarks Uraftgrupp AB</td>
<td>Sweden</td>
<td>+46 173 81000</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+46 173 55116</td>
</tr>
<tr>
<td>Magnus Arvidson</td>
<td>Swedish National. Testing &amp; Research</td>
<td>Sweden</td>
<td>+46(0) 33 165</td>
</tr>
<tr>
<td></td>
<td>Institut, SP</td>
<td></td>
<td>+46(0) 33 117759</td>
</tr>
<tr>
<td>Heinz Peter Berg</td>
<td>BIS</td>
<td>Germany</td>
<td>(05341) 225114</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(05341) 225225</td>
</tr>
<tr>
<td>Gőran Berghvist</td>
<td>Forsmarks Kaftgrupp AB</td>
<td>Sweden</td>
<td>+46 173 81000</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+46 173 55116</td>
</tr>
<tr>
<td>Lars I. Berglund</td>
<td>Swedish Nuclear Power Inspectorate</td>
<td>Sweden</td>
<td>+46 8 665 4436</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+46 8 661 9086</td>
</tr>
<tr>
<td>Beuchler</td>
<td>TÜV Nord</td>
<td>Germany</td>
<td>(040) 85572748</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(040) 85572116</td>
</tr>
<tr>
<td>Herbert Bittner</td>
<td>KWU</td>
<td>Germany</td>
<td>(09131) 1825..</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(09131) 182596</td>
</tr>
<tr>
<td>Laszlo Bokor</td>
<td>PAKS</td>
<td>Hungary</td>
<td>+36 75317931</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+36 1155132</td>
</tr>
<tr>
<td>Michael L. Brown</td>
<td>AEA</td>
<td>United</td>
<td>(0235) 46 4113</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Kingdom</td>
<td>(0235) 46 4136</td>
</tr>
<tr>
<td>Jean Chapus</td>
<td>EdF</td>
<td>France</td>
<td>+33 14922 8119</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+33 14922 8155</td>
</tr>
<tr>
<td>Jean-Pierre Clausner</td>
<td>OECD / NEA</td>
<td>France</td>
<td>(33-1) 4524 1054</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(33-1) 4524 1110</td>
</tr>
<tr>
<td>Bruno Covelli</td>
<td>TECOVA AG</td>
<td>Switzerland</td>
<td>(057) 227930</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(057) 229812</td>
</tr>
<tr>
<td>J. S. Cowley</td>
<td>Health &amp; Safety Executive</td>
<td>United</td>
<td>+44 519 51 4395</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Kingdom</td>
<td>+44 519 22 5980</td>
</tr>
<tr>
<td>Knut Frisch</td>
<td>KK Brunsbüttel</td>
<td>Germany</td>
<td>(04852) 892060</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(04852) 892009</td>
</tr>
<tr>
<td>P K Ghosh</td>
<td>AERB</td>
<td>India</td>
<td>+22 556 3218</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+22 556 2344</td>
</tr>
<tr>
<td>Enrico Gorza</td>
<td>Tractebel</td>
<td>Belgium</td>
<td>+32 2 773 8236</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+32 2 773 9900</td>
</tr>
<tr>
<td>Mikio Hada</td>
<td>Nupec</td>
<td>Japan</td>
<td>+ 81 3 54 70 5500</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+ 81 3 54 70 5524</td>
</tr>
<tr>
<td>R. J. Hadden</td>
<td>Health &amp; Safety Executive</td>
<td>United</td>
<td>+44 519 51 4395</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Kingdom</td>
<td>+44 519 22 5980</td>
</tr>
<tr>
<td>Guy Hebing</td>
<td>EdF.EPN.IN</td>
<td>France</td>
<td>+33(1) 4725 8952</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+33(1) 4725 8990</td>
</tr>
<tr>
<td>Gunnar Helleberg</td>
<td>Vattenfall AB</td>
<td>Sweden</td>
<td>+46 8 739 5418</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+46 8 877 469</td>
</tr>
<tr>
<td>Name</td>
<td>Organisation</td>
<td>Nationality</td>
<td>Phone/Fax</td>
</tr>
<tr>
<td>--------------------</td>
<td>---------------------------------------------------</td>
<td>-------------</td>
<td>----------------------------</td>
</tr>
<tr>
<td>Hero</td>
<td>TÜV Bayern/Sachsen</td>
<td>Germany</td>
<td>+49 89 57911479 +49 89 57911479</td>
</tr>
<tr>
<td>Heinz H. Hoffmann</td>
<td>ABB</td>
<td>Germany</td>
<td>(0621) 3882272 (0621) 3882390</td>
</tr>
<tr>
<td>Dietmar Hosser</td>
<td>TU Braunschweig</td>
<td>Germany</td>
<td>(0531) 3915441 (0531) 3914573</td>
</tr>
<tr>
<td>H. Ingason</td>
<td>Swedish National Testing &amp; Research Institut, SP</td>
<td>Sweden</td>
<td>+46(0) 33 16 ... +46(0) 33 117759</td>
</tr>
<tr>
<td>Goran Ingers</td>
<td>Sydsvenska Varme-kraft AB</td>
<td>Sweden</td>
<td>+46 0467 24000 +46 0467 75848</td>
</tr>
<tr>
<td>Sören Isaksson</td>
<td>Swedish National Testing &amp; Research Institut, SP</td>
<td>Sweden</td>
<td>+46(0) 33 165205 +46(0) 33 117759</td>
</tr>
<tr>
<td>Ales Janezic</td>
<td>Slovenian Nuclear Safety Administration</td>
<td>Slovenia</td>
<td>(386) 61 343 751 (386) 61 343 667</td>
</tr>
<tr>
<td>Fredrik Jörud</td>
<td>Sydkraft Konsult AB</td>
<td>Sweden</td>
<td>+46 40 25 6000 +46 40 25 6028</td>
</tr>
<tr>
<td>Maurice Kaercher</td>
<td>EdF</td>
<td>France</td>
<td>+33 7894 7629 +33 7894 4999</td>
</tr>
<tr>
<td>Helmut Karwat</td>
<td>TU München</td>
<td>Germany</td>
<td>(089) 32004128 (089) 32004299</td>
</tr>
<tr>
<td>Olavi Keski-Rahkonen</td>
<td>VTT</td>
<td>Finland</td>
<td>+358 0 456 4810 +358 0 456 4815</td>
</tr>
<tr>
<td>Klaus Kotthoff</td>
<td>GRS</td>
<td>Germany</td>
<td>(0221) 2068417 (0221) 2068442</td>
</tr>
<tr>
<td>Jean Claude Laborde</td>
<td>IPSN</td>
<td>France</td>
<td>(1) 6908 3777 (1) 6908 3680</td>
</tr>
<tr>
<td>Jocelyne Lacoue</td>
<td>IPSN</td>
<td>France</td>
<td>+33 1 46 54 91 70 +33 1 46 51 91 71</td>
</tr>
<tr>
<td>Penh Lamuth</td>
<td>IPSN</td>
<td>France</td>
<td>+33 1 46 54 9103 +33 1 46 51 91 71</td>
</tr>
<tr>
<td>Lars-Eric Larsson</td>
<td>STF</td>
<td>Sweden</td>
<td>+46 340 88321 +46 340 15602</td>
</tr>
<tr>
<td>Heinz Liemersdorf</td>
<td>GRS</td>
<td>Germany</td>
<td>(0221) 2068518 (0221) 2068442</td>
</tr>
<tr>
<td>Marie-Claude Lopez</td>
<td>IPSN</td>
<td>France</td>
<td>(33) 76884203 (33) 76885156</td>
</tr>
<tr>
<td>Patrick Madden</td>
<td>USNRC</td>
<td>USA</td>
<td>(301) 504 2854 (301) 504 2260</td>
</tr>
<tr>
<td>Tommy Magnusson</td>
<td>Ringhals / Vattenfall</td>
<td>Sweden</td>
<td>+46 340 67294 +46 340 68146</td>
</tr>
<tr>
<td>Name</td>
<td>Organisation</td>
<td>Nationality</td>
<td>Phone/Fax</td>
</tr>
<tr>
<td>-------------------------------</td>
<td>---------------------------------------------------</td>
<td>-------------</td>
<td>-------------------</td>
</tr>
<tr>
<td>Nobuo Maki</td>
<td>OECD / NEA</td>
<td>Japan</td>
<td>+33 1 4524 1067</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+33 1 4524 1110</td>
</tr>
<tr>
<td>Giuseppe Marella</td>
<td>ENEA-DISP</td>
<td>Italy</td>
<td>+39 6 5007 2124</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+39 6 5007 2044</td>
</tr>
<tr>
<td>Jouko Marttila</td>
<td>Finnish Center for Radiation &amp; Nuclear Safety</td>
<td>Finland</td>
<td>+358 0 7082 1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+358 0 7082 392</td>
</tr>
<tr>
<td>Wolfhard Meier</td>
<td>Minist. f. Umwelt Rheinland-Pfalz</td>
<td>Germany</td>
<td>(06131) 162123</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(06131) 164644</td>
</tr>
<tr>
<td>Hubert Mennessiez</td>
<td>DSIN</td>
<td>France</td>
<td>+33 1 4654 9204</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+33 1 4654 4362</td>
</tr>
<tr>
<td>Mª Dolores Morales Dorado</td>
<td>C.N. Almaraz</td>
<td>Spain</td>
<td>(1) 4314222</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(1) 4357310</td>
</tr>
<tr>
<td>Klaus Müller</td>
<td>KIK</td>
<td>Germany</td>
<td>(07247) 82 4343</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(07247) 82 4386</td>
</tr>
<tr>
<td>Tore Nilsson</td>
<td>Swedish Nuclear Power Inspectorate</td>
<td>Sweden</td>
<td>+46 8 665 4456</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+46 8 661 9086</td>
</tr>
<tr>
<td>Antti Norta</td>
<td>IVO</td>
<td>Finland</td>
<td>+358 0 508 2276</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+358 0 508 3407</td>
</tr>
<tr>
<td>Richard N. Oehlberg</td>
<td>EPRI</td>
<td>USA</td>
<td>+1 415 855 2082</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+1 415 855 1026</td>
</tr>
<tr>
<td>Hans Dieter Plum</td>
<td>TÜV Südwest</td>
<td>Germany</td>
<td>(0621) 395363</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(0621) 395591</td>
</tr>
<tr>
<td>Mani Pradhan</td>
<td>KTA-GS (BIS)</td>
<td></td>
<td>(05341) 225204</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(05341) 225105</td>
</tr>
<tr>
<td>Marina Röwekamp</td>
<td>GRS</td>
<td>Germany</td>
<td>(0221) 2068384</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(0221) 2068442</td>
</tr>
<tr>
<td>Francisco Javier Sanz Román</td>
<td>C.N. Almaraz</td>
<td>Spain</td>
<td>(1) 4314222</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(1) 4357310</td>
</tr>
<tr>
<td>Frank Sauer</td>
<td>Preussen Electra</td>
<td>Germany</td>
<td>(05111) 4390</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(05111) 4392375</td>
</tr>
<tr>
<td>Ulrich Schneider</td>
<td>TU Wien</td>
<td>Austria</td>
<td>+43 1 58801 3393</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+43 1 5056726</td>
</tr>
<tr>
<td>Uwe Sieberg</td>
<td>RWTÜV</td>
<td>Germany</td>
<td>(0201) 825 3258</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(0201) 825 3347</td>
</tr>
<tr>
<td>John G. Smyth</td>
<td>Council for Nuclear safety</td>
<td>South Africa</td>
<td>+27 12 663 5500</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+27 12 663 5513</td>
</tr>
<tr>
<td>Lothar Sonnenschein</td>
<td>RWTÜV</td>
<td>Germany</td>
<td>(0201) 825 3258</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(0201) 825 3347</td>
</tr>
<tr>
<td>Pierre Staeltert</td>
<td>D.T.V.K.I.</td>
<td>Belgium</td>
<td>+32 2 233 45 28</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+32 2 233 45 31</td>
</tr>
<tr>
<td>Kari Juhana Taivainen</td>
<td>Teollisuudan Voima Oy (TVO)</td>
<td>Finland</td>
<td>+358 0 3838 11</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+358 0383812109</td>
</tr>
<tr>
<td>Name</td>
<td>Organisation</td>
<td>Nationality</td>
<td>Phone/Fax</td>
</tr>
<tr>
<td>--------------------</td>
<td>-------------------------------</td>
<td>-------------</td>
<td>----------------------------</td>
</tr>
<tr>
<td>Gerard Tenback</td>
<td>N.V. GKN</td>
<td>Netherland</td>
<td>+31 8885 8217</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+31*8885 2128</td>
</tr>
<tr>
<td>Werner Töpfer</td>
<td>RSK-GS, BfS</td>
<td>Germany</td>
<td>(0228) 3053756</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(0228) 670388</td>
</tr>
<tr>
<td>L. Tuvesson</td>
<td>ABB Atom</td>
<td>Sweden</td>
<td>+46</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+46</td>
</tr>
<tr>
<td>Andre Vandevalle</td>
<td>AIB Doel/Tihange Inspection Branch</td>
<td>Belgium</td>
<td>+32 2 536 8211</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+32 2 536 8585</td>
</tr>
<tr>
<td>Jaques Vendel</td>
<td>IPSN</td>
<td>France</td>
<td>(1) 6908 3777</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(1) 6908 3680</td>
</tr>
<tr>
<td>Andrea Waldorf</td>
<td>RSK-GS, BfS</td>
<td>Germany</td>
<td>(0228) 3053754</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(0228) 670388</td>
</tr>
<tr>
<td>Ralf-Michael Zander</td>
<td>KRB</td>
<td>Germany</td>
<td>(08224) 782249</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>(08224) 782900</td>
</tr>
<tr>
<td>Wanli Zhong</td>
<td>IAEA</td>
<td>China</td>
<td>+43 1 2360 2517</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>+43 1 2360 234564</td>
</tr>
</tbody>
</table>